Duke Power
A Duke Energy Company

Gary R. Peterson
Vice President

Duke Power
Catawba Nuclear Station
4800 Concord Road
York, SC 29745
(803) 831-4251 OFFICE
(803) 831-3221 FAX

April 1, 2001

U. S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Duke Energy Corporation
Catawba Nuclear Station Unit 1 and Unit 2
Docket Numbers 50-413 and 50-414
2000 10CFR50.59 Report

Attached please find a report containing a brief description of changes, tests, and experiments, including a summary of the safety evaluation of each, for Catawba Nuclear Station Units 1 and 2 during 2000. This report is being submitted per the provisions of 10CFR50.59(b)(2) and 10CFR50.4.

Questions regarding this report should be directed to J. W. Glenn at (803) 831-3051.

Sincerely,

G. R. Peterson

G. R. Peterson

Attachment

IE47

xc:

L. A. Reyes
U. S. Nuclear Regulatory Commission
Regional Administrator, Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30303

C. P. Patel
NRC Senior Project Manager (CNS)
U. S. Nuclear Regulatory Commission
Mail Stop O-8H12
Washington, DC 20555-0001

D. J. Roberts
Senior Resident Inspector (CNS)
U. S. Nuclear Regulatory Commission
Catawba Nuclear Site

Catawba Nuclear Station

Units 1 and 2

2000 10CFR50.59 Report

April 1, 2001

This report consists of a summary of changes, tests, and experiments, including a summary of the safety evaluation of each, for Catawba Nuclear Station, Units 1 and 2, for 2000. The entries are organized by the type of activity being evaluated in the following order:

205 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-08567, Delete Valves 1NV-271 and 1NV-291 and replace with blank flanges.

Description: Modification CE-08567 replaces the Unit 1 alternate seal injection flow path valves 1NV-271 and 1NV-291 with blank flanges to preclude an undesirable leak path.

A review performed of this modification package and its original 10CFR50.59 Evaluation has concluded that an inadequate evaluation was performed. This evaluation will replace the original evaluation performed for modification CE-08567.

Five basic issues were addressed in this evaluation:

1. The effect on Reactor Coolant Pump operation.
2. Loss of seal injection.
3. Loss of seal injection and loss of Component Cooling Water, a SSF event.
4. Seismic/Structural integrity.
5. High-energy line breaks.

Evaluation: The two valves, 1NV-271 and 1NV-291 are used to isolate an alternate seal injection line. The replacement of these valves with blank flanges will maintain the integrity of the normal seal injection line supplied by a Centrifugal Charging Pump. This line is not a UFSAR accident initiator nor is it used in any existing procedure to mitigate any design basis accidents. Reactor Coolant Pump seals are still protected by the normal method of seal injection from the Centrifugal Charging Pump. The normal method of seal injection is assured by safety related valves that can be relied upon to go to their design position in the event of an accident. In the event of a failure of forced seal injection from the Centrifugal Charging Pumps, the Reactor Coolant Pumps have a thermal barrier that would cool the Reactor Coolant System water as it back-flowed through the seals thus providing the same seal cooling as the forced seal injection. Since this alternate seal injection line is not described in the UFSAR and none of the Operations procedures use this line to mitigate an accident there is no increase in the probability of occurrence of an accident previously evaluated in the UFSAR. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. No UFSAR changes are required.

42 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-10284, Revise Filter Descriptions for Supply Filters

Description: Minor Modification CE-10284 makes revisions to Vendor Air Handling Unit drawings, Auxiliary Building Ventilation System, Fuel Pool Ventilation System and Diesel Building Ventilation System Design Basis Documents and UFSAR Section 9.4. The purpose of these revisions is to delete descriptions concerning filter efficiencies based on old ASHRAE test methods and to change the description of the filters to refer to them as 'particulate' filters. The vendor filter drawing will have TRI-DEK brand filters added as well as revising the existing notes to add fastening options for stiffeners.

Evaluation: There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. UFSAR Section 9.4. will be revised to remove references to particulate filter efficiencies and old ASHRAE Standard Test descriptions. Changes to the manufacturer's drawings were made to delete reference to "bag" filter sections and to refer to these sections as "filter" sections. Some drawings on the Auxiliary Building Ventilation Supply Units and Fuel Pool Supply Units are revised to reflect the absence of downstream final-filters. In the UFSAR Chapter 9 Sections and in Auxiliary Building Ventilation System, Fuel Pool Ventilation System and Diesel Building Ventilation System Design Basis Documents, the description of the filters currently state an efficiency rating based on an old ASHRAE Test method. This is being removed and the reference to "pre-filters" and "final filters" is being changed to describe them as particulate filters. The old ASHRAE test method is no longer used to test efficiencies of filters, and the efficiency ratings are not of particular significance to any safety or reliability issue in the plant. The filters need only meet current commercial standards for efficiency that ensure appropriate air quality in the building and performance information will not be re-instated in the UFSAR since it is not a critical factor. Notes are being modified on the filter vendor data sheet in the Minor Modification that will allow more flexibility in adding stiffeners in the filter racks. These racks are needed to hold the filters in place and prevent them from bowing and allowing un-filtered air from going through the air handling units. Additional brands of filters are being added for use in the supply units that resolve past problems with air being able to bypass the pre-filters. The new panel type filters are designed to help eliminate unfiltered air bypass. In some cases the current size filters will be replaced with deeper filters. These deeper "cube" filters will last longer and are a new design that will filter more from the air.

130 **Type:** Minor Modification **Unit:** 2

Title: Minor Modification CE-10334, Corrections to the Station Air System Flow Diagram

Description: UFSAR Figure 9-250 is the flow diagram for the Station Air System. This modification will correct the field locations for valves 2VS-400, 2VS-401, 2VS-402, 2VS-403, 2VS-404 and 2VS-405, as listed on flow diagram CN-1605-2.3. Corrections will also be made to the Equipment Database.

Evaluation: There is no unreviewed safety question asssociated with revising UFSAR Figure 9-250. This is an editorial change. There are no actual changes to the plant. This revision has no effect on the probability or consequences of accidents evaluated in the UFSAR. No Technical Specification changes are required. UFSAR Figure 9-250 will be revised.

103 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-10444, Revise Containment Spray Pump vent configuration

Description: Modification CE-10444 will revise the vent piping for valves 1NS27 and 1NS28 on Containment Spray Pump 1A and 1B respectively. The current configuration has the vent in a vertical position making it difficult to vent the pumps. This modification will place the vent in a horizontal position making it more conducive to performing associated activities with the Containment Spray Pumps.

Evaluation: The reconfiguration of the vents will utilize the existing valves and associated piping. This modification will simply provide a more efficient and convenient method for venting the Containment Spray Pump seals. The affected portions of the on Containment Spray System will continue to function as described in the UFSAR and the design basis specification. ECCS operation will not be affected by reconfiguration of these vent lines.

This modification does not involve any Unreviewed Safety Questions. No changes to the Technical Specifications are required. UFSAR changes are required for Figure 6-109 (a Piping Flow Diagram).

55 **Type:** Minor Modification **Unit:** 2

Title: Minor Modification CE-10445, Revise Containment Spray Pump vent configuration

Description: Minor Modification CE-10445 will revise the vent piping for valve 2NS-28 on Containment Spray Pump 2B. The current configiuration has the vent in a vertical position making it difficult to vent the pump. This modification will place the vent in a horizontal position which will allow for easier performance of activities associated with the pump.

Evaluation: There are no unreviewed safety questions associated with this modification. The reconfiguration of the vent will utilize the existing valves and piping. The modification will provide a more efficient and convenient method for venting the Containment Spray Pump seals. The affected portion of the Containment Spray System will continue to function as currently described in the UFSAR. Emergency Core Cooling System performance will not be affected by this modification. The modification will have no effect on the probability or consequrences of accidents described in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required.

108 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-10476, Correction of document inconsistencies

Description: UFSAR Figure 10-4 will be revised to reflect the following changes, as incorporated in the Hydrogen Bulk Storage System Flow Diagram and the Hydrogen Bulk Storage System Description. These documents have been updated to reflect "Good Industry Practices" for Bulk Hydrogen Gas storage. That is, hydrogen gas, when pressurized and in contact with air, is self-combustive by nature.

As a result of this review the following actions were implemented:

1. A safer alignment was achieved by restricting the open cylinder(s) to the minimum required to meet plant needs.
2. A cylinder rotation was established controlled through the normal system operating procedure, such that each cylinder and the corresponding equipment will be exercised.
3. Specific valve alignments for 1GS60, 1GS21, and 1GS28 were made to support the safer alignment.
4. More guidance was provided for purging hydrogen in support of planned and corrective maintenance.
5. The Equipment Database was updated with the regulator setpoints for valve 1GS17 and 1GS22.
6. The Oxygen System and Bulk Hydrogen System Descriptions were separated into two distinct documents.
7. UFSAR Figure 10-4 was revised to reflect the current system alignment and correct the title of Figure 10-4.

Evaluation: There is no Unreviewed Safety Question associated with this modification. There has been no increase in the probability or consequences of accidents evaluated in the UFSAR. No changes to the Technical Specifications are required. UFSAR Figure 10-4 will be revised.

6 **Type:** Minor Modification **Unit:** 2

Title: Minor Modification CE-10612, replace valve 2CF-126 with a new valve

Description: Minor Modification CE-10612 will replace valve 2CF-126 (Item Number 06J-238) with a new valve (Item Number 06J-335). Presently this valve is a one inch Y-type globe valve used in a drain application. The replacement valve is a one inch gate valve which is suitable for this application. All affected drawings will be revised to reflect this new information.

The Main Feedwater System supplies feedwater to the four steam generators at the temperature, pressure, and flow required to maintain the proper steam generator water level. For both Unit 1 and Unit 2, Steam Generator water level is proportional to reactor power output. The Main Feedwater System serves no safety function (with the exception of containment isolation integrity) and is, therefore, primarily classified as non-nuclear safety related. Valve 2CF-126 is a drain valve for the 2A Steam Generator.

Evaluation: The new valve has been evaluated to be a suitable replacement. This modification will have no effect on the probability or consequences of accidents analyzed in the UFSAR. There are no Unreviewed Safety Questions associated with this minor modification. Flow diagram CN 2591-01.01 (not a UFSAR Figure) will be revised to show the new valve design. No Technical Specification changes are required. No UFSAR changes are required.

160 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-10737, Perform various motor operator replacements and MOV testing to support NRC Generic Letters 89-10 and 96-05

Description: NRC Generic Letters 89-10 and 96-05 require a higher level of operability determination, maintenance, and surveillance of critical motor operated valves (MOVs). The following valves are affected by this Minor Modification: 1BB-001, 1BB-013, 1BB-021B, 1BB-061B, 1 KC-394A, 1 ND-065B, 1NI-054A, 1NI-065B, 1NI-076A, 1NI-088B, 1NI-103A, 1NI-121A, 1NI-135B, 1NI-150B, 1NI-152B, 1NI-162A, 1NI-184B, 1NI-185A, 1NI-334B, 1NI-438A, 1NM-003A, 1NS-003B, 1NS-015B, 1NS-032A, 1NS-038B, 1NS-043A, 1NV-001, 1NV-51B, 1NV-055A, 1NV-077A, 1NV-188A, 1NV-203A, 1RN-232A, 1RN-0292B, 1SV-025B, 1SV-026B, 1SV-027A, 1SV-028A, 1WL-805A, 1WL-867A and 1WL-869B. Meeting the requirements of Generic Letters 89-10 and 96-05 will require revisions to CNM-1205.00-1997.001, "Torque Switch Setting Sheets," which serves as the source for valve testing and set-up data. New thrust or torque set-up windows are being established to increase each MOV's margin for operation.

In addition, new actuators are being installed on 1ND-065B, 1NI-150B, 1NS-015B, 1NS-032A, 1NV-015B, 1SV-025B, 1SV-026B, 1WL-805A and a new motor is being installed on 1N1-065B. The actuators on 1SV-027A and 1SV-028A are being modified to achieve a higher thrust limit on the actuator. These modifications are also being performed to increase each MOV's margin for operation.

Evaluation: There is no change in the operation of the valves or associated systems due to this modification. The valves will function when called upon just as they did prior to this modification. No new failure modes have been created as a result of this modification.

Minor Modification CE-10737 and Variation Notice VN-10737B do not involve an Unreviewed Safety Question. UFSAR changes are required to Unit One flow diagrams affected by this modification. No Technical Specification changes are required.

120 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-10788, Install a replacement fuseholder for device MD46 on Main Control Board 2MC1

Description: Modification CE-10788 will replace an obsolete fuse holder (Bussman Part Number 4575) with the replacement model recommended by the manufacturer (Bussman Part Number BM6032SQ) in Main Control Board 2MC1. This fuse holder transfers AC power from terminal board MD6 to the 48 VDC power supply. The fuseholder is associated with the Digital Rod Position Indication System. This item does not serve a nuclear safety related function.

Evaluation: The existing mounting for the fuse holder will also be revised because of the different characteristics between the two models. The mounting is seismic and a review was performed to address the seismic issue. The replacement of an obsolete fuse holder with the currently available, manufacturer recommended replacement does not represent an unreviewed safety question. The modification has no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required.

105 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-10811, Correct errors on Instrument Air System flow diagrams and reposition a manual valve from open to closed

Description: Modification CE-10811 will update UFSAR Figures 9-229, 9-235, 9-236, 9-244, and 9-245. The figures will be changed to reflect the current plant equipment and identification. The changes include: 1) changing the normal valve position of valve 1VI-681 from open to closed. This valve is located on a capped "dead-end" tee. This change will make no difference to the individual component operation or system operation. Closing this valve is utilizing "Good System Practices" for keeping valves closed which are located on "dead-end" capped pipe tees. They do not supply air to any components. This change provides a double isolation against air leaks. In addition several editorial drawing errors will be corrected. The Instrument Air System will continue to operate as designed.

Evaluation: There are no Unreviewed Safety Questions associated with change. Correcting drawings has no effect on the probability or consequences of accidents evaluated in the UFSAR. No changes to the Technical Specifications are required. UFSAR Figures 9-229, 9-235, 9-236, 9-244, and 9-245 will be revised.

9 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-10831, Isolate valve 1(2)CM-33 and 1,2CM-127 flowpaths to the Upper Surge Tasnk

Description: This modification addresses:

1) Isolation of valve 1(2)CM-33, which identified a need to evaluate the potential Adverse System Interaction involving valve 1(2)CM-33.
2) The isolation of valve 1(2)CM-127 which identified a need to evaluate the potential Single Failure of the Non-Nuclear Safety Related valves 1(2)CM-127.

Valves 1(2)CM-33 provide mini-flow protection for the non-nuclear safety related Condenser Hotwell Pumps and means of Hotwell inventory control on high Hotwell level. Valves 1(2)CM-33 have the potential to supply hot water to the Auxiliary Feedwater System suction in excess of the Auxiliary Feedwater System piping design temperature in the event of a loss of condenser vacuum accident. The Auxiliary Feedwater System is not required for mitigation of the loss of vacuum accident, however, it is required to provide core cooling in the long term following the accident. The potential Adverse System Interaction could render the Auxiliary Feedwater System inoperable.

Valves 1(2)CM-127 are used during normal full power operation to provide mini-flow protection for the non-nuclear safety related Condensate Booster Pumps. During a Unit 1 power reduction, valve 1CM-127 opened as a result of an incorrect setpoint at approximately 50% power. As a result of the valve opening, hot water was admitted to the Upper Surge Tank. The Upper Surge Tank is currently the preferred non-nuclear safety related source for the Auxiliary Feedwater System and the elevated temperatures resulted in declaring all three the Auxiliary Feedwater System Pumps inoperable.

Evaluation: This evaluation address the effects of isolation of the flow paths on the Condenser Hotwell Pumps, the Condensate Booster Pumps, and the Secondary Systems. The evaluation states that during normal operation minimum flow protection can be provided by the Main Feedwater Pump Recirculation valves, and that this change along with a change in power level for placing the C Heater Drain Pumps in service does not materially change operation of the system from that described in the UFSAR. Additionally, other means can be used to reduce Hotwell inventory if required during normal operation.

There are no Unreviewed Safety Questions associated with this modification. No changes to Technical Specifications are required. A change is required to UFSAR Sections 10.4.1 and 10.4.7.5.2.

56 **Type:** Minor Modification **Unit:** 2

Title: Minor Modification CE-10938, Replace valves 2ZP-020 and 2ZP-025 with a new valve type.

Description: This Minor Modification will replace valves 2ZP-020 and 2ZP-025 (Item Number 06J-601) with new valve with an Item Number of DSV-031. Presently these valves are one inch Y-Type Globe valves. The replacement valves are one inch ball valves that are suitable for these applications. All affected drawings will be revised to reflect this new information.

The Vacuum Priming System is used to continuously remove air from cooling water lines, condenser waterboxes and other heat exchangers that operate under a vacuum on the cooling water side. Valves 2ZP-020 and 2ZP-025 are the 2A1and 2A2 Main Condenser Outlet Waterbox Priming Valve outlets.

Evaluation: There is no Unreviewed Safety Question associated with this minor modification. Flow diagram CN 1598-03.01 (not a UFSAR Figure) will be revised to show the new valve designs. No Technical Specification changes are required. No UFSAR changes are required.

10 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-10956, Replace valves 1VG-5, 1VG-6, 1VG-7, 1VG-8, 1VG-49 ,1VG-50, 1VG-51, 1VG-52,

Description: Valves 1VG-5, 1VG-6, 1VG-7, 1VG-8, 1VG-49 ,1VG-50, 1VG-51, and 1VG-52 are currently carbon steel Kerotest lift check valves. The valves are experiencing corrosion related problems and are causing maintenance and operational concerns. All eight of these valves revealed corrosion when inspected. These valves will be replaced with stainless steel swing check valves which were identified by Engineering as a suitable replacement.

Evaluation: There are no unreviewed safety questions associated with this modification. The new valves were evaluated to be a suitable replacement for the old valves. No Technical Specification changes are required. Changes are required for UFSAR Figures 9-183 and 9-184.

63 Type: Minor Modification **Unit:** 2

Title: Minor Modification CE-10986, Replace Valve 2RN46

Description: Minor Modification CE-10986 will replace valve 2RN46 (Item Number 02B-331) with new valve (Item Number 05D-709). Valve 2RN46 is the 2A Nuclear Service Water Strainer Drain Isolation Valve. Presently this valve is a three inch T-Type Globe valve with internal damage. It will be replaced with a three inch Ball valve, which will continue to serve the same function. All affected drawings will be revised to reflect this new information.

Evaluation: There are no Unreviewed Safety Questions associated with this modification. Replacement of valve 2RN46 with an equivalent component is a maintenance activity that is not addressed in the UFSAR and is not a significant plant change that would require inclusion in the UFSAR. This Minor Modification involves a one-for-one component replacement/improvement. The overall gain should be in enhanced system operability and availability. The accidents evaluated in the UFSAR do not include this valve in any design basis, therefore an increased probability of an accident will not occur due to the replacement of 2RN46. The new valve will serve the same function as the old valve,therefore there will be no increased consequences of an accident or equipment malfunction. The design basis of this minor modification is consistent with the previous design basis of the Nuclear Service Water System and will not create any new type of accidents or malfunctions. No Technical Specification changes are required. No UFSAR changes are required.

64 Type: Minor Modification **Unit:** 2

Title: Minor Modification CE-10991, Replace Valve 2ZP42

Description: Modification CE-10991 will replace valve 2ZP42 (Item Number CMV-662) with new Item Number DSV-031. Presently this valve is a one inch T-Type Globe valve. The replacement valve is a one inch Ball valve that is suitable for this application. All affected drawings will be revised to reflect this new information.

The Vacuum Priming System is used to continuously remove air from cooling water fines, condenser waterboxes and other heat exchangers that operate under a vacuum on the cooling water side. Valve 2ZP42 is the 2A and 2B Hydrogen Coolers Priming Valve Outlet.

Evaluation: The replacement valve for 2ZP42 will continue to provide all requirements necessary for plant operation and safety once it is replaced. There is no Unreviewed Safety Question associated with this valve change-out. Flow diagram CN 1598-03.01 (not a UFSAR Figure) will be revised to show the new valve designs. No Technical Specification changes are required. No UFSAR changes are required.

77 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-10993

Description: Minor Modification CE-10993 will allow unrestricted use of valve NV-849 for any flow condition within the original design of the Chemical and Volume Control System Letdown Line.

Catawba Nuclear Station discontinued using the Chemical and Volume Control System variable letdown orifice valves 1(2)NV-849 upon discovering that the valve and associated piping were experiencing high vibration.

Valve NV-849 had been qualified following the identification of the vibration problem for short term use to warm the letdown line in the case of initiating letdown or the re-initiation of letdown following a loss of letdown. The valve was to be used for this purpose only when Reactor Coolant System pressure is greater than 350 psig. When the Reactor Coolant System pressure was less than 350 psig, NV-849 could be used at all times.

Subsequently, the piping was reconfigured using butt welds instead of the original socket welds, which significantly reduced the vibration in the piping. Valve trim cages were replaced with a newer version less susceptible to producing vibration. The letdown lines were retested and the vibration levels measured. Data gathered during the testing of the Unit 1 letdown orifices determined that the variable orifice valve 1NV-849 was acceptable to use in any combination of flow orifices and for any allowable flow except in the 80 to 110 gpm range when operating by itself. Evaluation of the data reveals that the vibration levels measured at 110 gpm using valve 1NV-849 by itself were probably not valid, but this flow restriction will remain in effect until the test can be repeated. Unit 2 data supports unconditional use of valve 2NV-849 for any allowable flow conditions using any combination of different orifices. Thus Unit 2 can be returned to a condition within the original design of the letdown line. 2NV-849 can be placed in service for as long as desired. The travel stops for valve NV-849 (on both units) were set up for a maximum allowable flow of 110 gpm at normal Reactor Coolant System operating temperature and pressure. This evaluation supports the use of valve NV-849 as originally designed and described in the UFSAR with the exception of using 1NV-849 by itself at flows greater than 80 gpm.

Several UFSAR sections describe the function of valve NV-849. NV-849 is not safety-related nor is it used to mitigate any of the accident scenarios described in the UFSAR. The flow path through NV-849 is isolated by safety-related equipment that operates on safety-related signals.

There is a possibility of increasing the allowable letdown flow for use in outages when the amount of time required to cleanup the Reactor Coolant System is critical. An evaluation by Engineering concluded that increasing the maximum allowable letdown flow to 150 gpm while on the Residual Heat Removal system is acceptable.

The evaluation was conducted by reviewing each component in the letdown flow path and determining if it was acceptable to use at 150 gpm. The manufacturer was contacted or an

Engineering evaluation performed to upgrade the components with flowrates less than 150 gpm to the new 150 gpm allowable flow. The results are documented in calculation CNC- 1223.04-00-0082 (Documentation for the Acceptability of up to 150 gpm Letdown Flow). The components evaluated were: Piping, Valves, Heat Exchanger, Mixed Bed Demineralizer Inlet Strainers, Mixed Bed Demineralizer, Reactor Coolant System Filters, Volume Control Tank (VCT) , Spray Nozzle, and Control Board Instrumentation

Procedures may be developed to operate the letdown at up to 150-gpm letdown during times of unit shutdown while on the Residual Heat Removal System. Currently the procedures for this evolution operate the system at a maximum flow of 120 gpm letdown. The Reactor Coolant System would be borated to a Shutdown Margin (SDM) for the coolest temperature expected if not for mode 6 SDM. The current procedures in effect for use of the Mixed Bed Demineralizer call operator attention to the level of boron currently existing in a Mixed Bed Demineralizer about to be placed in service. Procedure steps exist to property borate a Mixed Bed Demineralizer if it is not already boron saturated for the current conditions.

The piping and valves that are used in the letdown flowpath are currently purchased and built to Duke Energy Class "B" and "C" for radioactive systems. Thus the piping is designed for seismic stresses that may be encountered over the life of the station. The letdown operation is well within the design parameters given for the piping and valves. The letdown flow path is already designed to accommodate postulated station blackouts as the valves required to maintain Reactor Coolant System inventory will either fail closed or can be closed using emergency backup power available to electric motor operated valves.

There is adequate heat removal capability in the letdown flowpath to ensure that the liquid temperature does not exceed the maximum allowable temperature for the Mixed Bed Demineralizers. The automatic functions protecting the Mixed Bed Demineralizers are still in place and would automatically divert any liquid greater than the 138 degrees F around the Mixed Bed Demineralizers by repositioning valve NV-153. Control Room alarms are in place in the control room to alert an operator of a high temperature existing in the letdown liquid.

Currently running letdown flow at greater than 128 gpm will provide an annunciator alarm in the control room. Also if 150-gpm letdown is used, it is reasonable to expect that the high charging flow annunciator in the control room would also be in alarm. This has been discussed with Operations and will be addressed via procedure changes. This is not considered a significant deviation to the annunciators since it is only used in outages and can be countered with temporary Operator Aid Computer alarms or increased surveillance as Operations deems appropriate. The annunciator alarms will be able to perform their intended functions for normal plant operations.

Evaluation: The unconditional use of valve NV-849 is not an accident initiator as described in the UFSAR. This valve was included for unconditional use in the original plant design. This minor modification is to return this valve to its original purpose. NV-849 is not a safety-related valve. Its flowpath is isolated by safety related components in the event of an accident.

The use of the Residual Heat Removal System/Chemical and Volume Control System

letdown flowpath is only briefly covered in the SER and UFSAR. A failure in this line is not credited since it is only used at times of low energy in the Reactor Coolant System. The letdown flowpath is not considered to be an accident initiator.

The use of the standby Mixed Bed Demineralizers for lithium removal is described in a change to the UFSAR submitted in January of 2000. The Mixed Bed Demineralizers are not described as accident initiators in the UFSAR. The purification portion of the system is isolated in accident conditions.

Valve NV-849 is used to letdown from the Reactor Coolant System for chemical and volume control. This valve is isolated in any accident scenario and does not play a part in mitigating any accidents. A failure of NV-849 would cause the valve to fail open against its travel stops. The travel stops have been set to limit the maximum flow to 110 gpm. This is within the allowable 120-gpm design rating of the letdown line.

The allowable increased flow rating to 150-gpm of the letdown flowpath can only affect the inventory of the Reactor Coolant System and the ability of the Residual Heat Removal System to perform its function. Inventory in the Reactor Coolant System is protected by being able to isolate the letdown flowpath via emergency power supplied electric motor operated valves or by air operated valves that fail to the closed position. The residual heat removal function of the Residual Heat Removal System used during outages is not affected since letdown is bled off the main line after the liquid is cooled by the Residual Heat Removal Heat Exchanger. Since this flow path is used only in modes 5, 6, or no mode, flashing will not be a problem since the Reactor Coolant System is less than 200 degrees F. Again, this flow path can be isolated under the given accident or design basis events given in the UFSAR. This change in allowable flow does not affect the ability of safety related equipment to perform any given safety functions.

The changes made to the standby Mixed Bed Demineralizers for lithium removal does not in any way affect any equipment used to mitigate accidents or design basis events. The Mixed Bed Demineralizers are used for purification of the Reactor Coolant System and are isolated during accidents..

There is no unreviewed safety question associated with this modification. The modification has no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are reqiuired. No UFSAR changes are required.

208 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-11005, Changes to the Catawba Nuclear Station Lubrication Manual, UFSAR Section 6.2.1.1.2, Technical Specification Bases, and HVAC Documentation

Description: Controlled documentation for Containment Air Addition/Release System blowers lists the model numbers as CB-80 series, CB-80, or CB-86. Model CB-86 blowers have been ordered and installed in the plant. Drawing CNM 1207.00-10 is a performance curve for the model CB-80 blower. This modification creates drawing CNM 1207.00-10 Sheet 2 to add the performance curve drawing for model CB-86 blowers. There is only a minimal difference in the performance of the two types of blowers. UFSAR Section 6.2.1.1.2 and Technical Specification Bases 3.6.4 make a statement that the Containment Air Addition/Release System blowers are capable of developing a maximum negative pressure of 2.8 psig. This is true of the model CB-80 blowers but the model CB-86 can develop a higher negative pressure. Because both models are in use, the UFSAR and Technical Specification Bases 3.6.4 will be changed to indicate that the blowers can develop a negative pressure beyond containment design limits if allowed to run unchecked. Multiple administrative controls exist to prevent this from happening. The reference to a specific maximum developed negative pressure will be removed.

Evaluation: The Containment Air Addition/Release System is discussed in UFSAR Sections 6.2.1.1.2 and 9.5.10. Section 9.5.10 covers the Design Basis, System Description, Safety Evaluation, and Test and Inspection Requirements. The Containment Air Addition/Release System is designed to provide a means of controlling Containment pressure between 0.3 psig and -0.1 psig with the plant in Mode 1 through Mode 4 to comply with Technical Specification 3.6.4. The system serves no safety related function. The system operates by removing air from upper containment, filtering it, and releasing it through the unit vent. The Containment Air Addition/Release System air release fans that are located on the outlet side of the filters induce flow in the system. Per the UFSAR, the system is designed to automatically isolate valve 1,2VQ10 and shut off when containment pressure reaches 0.0 psig (UFSAR 9.5.10.2), on a unit vent radiation monitor alarm (UFSAR 9.5.10.2), or upon an ESF actuation (UFSAR 9.5.10.3). This modification involves only the non-safety related portion of the Containment Air Addition/Release System which is not involved in any accident scenarios and will not affect any other systems, structures, or components important for the safe shutdown of the plant. No accident analysis parameters will be affected and no margin of safety will be changed.

There are no unreviewed safety questions associated with this UFSAR Revision. No Technical Specification changes are required. A change is required for UFSAR Section 6.2.1.1.2

195 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-11009, Delete Sample Blower Data Sheet and Backup Documentation

Description: During an Accuracy Review of UFSAR Chapter 6 it was noted that the Containment Hydrogen Sample Blower design data was still included in UFSAR Table 6-81 and UFSAR Sections 6.2.5.3.2, 6.2.5.4.2 and 6.2.5.5. This component was removed by Design Change Authorization CN-1(2)-M 1132 in 1982, and replaced with the Post Accident Containment Sample (PACS) panel. The Sample Blower was functionally replaced with the PACS panel. Minor Modification CE-11009 was written to delete the Duke data sheet and supporting calculation for the Sample Blower and make the associated UFSAR changes.

Evaluation: There is no unreviewed safety question associated with this minor modification and UFSAR change. Revising the UFSAR to remove outdated information has no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. UFSAR Sections 6.2.5.3.2, 6.2.5.4.2 and 6.2.5.5 and UFSAR Table 6-81 will be revised.

209 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-11024, Remove the straightening section, honeycomb, from the Auxiliary Building Ventilation System air flow monitor 1ABUX-AFMD-1.

Description: This modification will remove the straightening section honeycomb from the Auxiliary Building Ventilation System air flow monitor 1ABUX-AFMD-1. The honeycomb has small openings, which straighten the air, and these openings are getting blocked with dirt and debris. After removal of the straightening section the air flow monitor will continue to perform its design function of monitoring air flow rate. After removal of the straightening section the seismic integrity of the air flow monitor will be maintained as a ducting enclosure for the passage of air during both normal and accident conditions. The Auxiliary Building Ventilation System filtered exhaust will be performance tested after this modification to ensure that it continues to meet air flow requirements of Technical Specification 5.5.11 "Ventilation Filter Test Program". No new failure modes have been identified with implementation of this modification.

Evaluation: There are no unreviewed safety questions associated with this UFSAR Revision. These air flow monitors are not required to function during an accident condition. The air flow monitor will maintain its structural integrity as a portion of the system ducting. No Technical Specification changes are required. No UFSAR changes are required.

210 **Type:** Minor Modification **Unit:** 2

Title: Minor Modification CE-11034, Remove the straightening section, honeycomb, from the Auxiliary Building Ventilation System air flow monitor 2ABUX-AFMD-1.

Description: This modification will remove the straightening section honeycomb from the Auxiliary Building Ventilation System air flow monitor 2ABUX-AFMD-1. The honeycomb has small openings, which straighten the air, and these openings are getting blocked with dirt and debris. After removal of the straightening section the air flow monitor will continue to perform its design function of monitoring air flow rate. After removal of the straightening section the seismic integrity of the air flow monitor will be maintained as a ducting enclosure for the passage of air during both normal and accident conditions. The Auxiliary Building Ventilation System filtered exhaust will be performance tested after this modification to ensure that it continues to meet air flow requirements of Technical Specification 5.5.11 " Ventilation Filter Test Program". No new failure modes have been identified with implementation of this modification.

Evaluation: There are no unreviewed safety questions associated with this UFSAR Revision. These air flow monitors are not required to function during an accident condition. The air flow monitor will maintain its structural integrity as a portion of the system ducting. No Technical Specification changes are required. No UFSAR changes are required.

97 **Type:** Minor Modification **Unit:** 2

Title: Minor Modification CE-61388, Install a new roof on the ground level pad located above the electrical equipment and penetration rooms

Description: Minor Modification CE-61388 will allow a contractor to install roofing on the Auxiliary Building ground level pad, located near the Unit 2 reactor building. Any remaining old roofing will be removed. The new roof will consist of a vented base sheet, two layers of smooth bituminous membrane applied with adhesive, and one layer of granular surfaced membrane all covered with a protective sheet and two inch pavers. This work includes new flashing and accessories. This area is important to the safe operation of the station.

Evaluation: This modification does not make any changes to a system or support structure as described in the UFSAR. The building roofs are not initiators of any accidents evaluated in the UFSAR. The modification will not change the function of any equipment related to safety. During implementation, it would be possible to degrade the carbon beds on the control room ventilation system if solvent fumes from the roofing process were allowed to be drawn into the control room ventilation system intake for unit 2. The filters have been evaluated for worst case degradation from the solvents used to perform roof work, and will remain operable. Precautions will be taken to minimize the probability of solvent contamination of the carbon beds. The roof contractor will be directed to not allow loose materials to be drawn into the intake, as this may damage the prefilters, and require their early replacement. Technical Specification Surveillance Requirement 3.7.10.2 requires that samples from the carbon beds in the filters be tested after solvents are released in any zone that communicates with a ventilation filter system. The roofing work does not directly communicate with the control room ventilation system, but the intake is located approximately four feet above the area to be roofed, so it is possible for fumes to be drawn into the system. Conservative decision making requires that SR 3.7.10.2 be utilized following completion of roof installation.

The configuration change to the roof will improve its function and extend its useful life. The UFSAR does not evaluate any accidents caused by an unsatisfactory performance of roofing.

There are no Unreviewed Safety Questions associated with this modification. No Technical Specification changes are required. No UFSAR changes are required.

203 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-61436, Replace vacuum switches on Radiation Monitors with mass flow computers

Description: Minor Modification CNCE-61436 changes the sample flow instrumentation on the following Shared and Unit 1 Process Radiation Monitors (EMFs):

EMF41	Auxiliary Building Single Range Beta Monitor
EMF43A and EMF43B	Control Room Air Intake Monitors
1EMF33	Condenser Air Ejector Exhaust Monitor
1EMF42	Fuel Building Ventilation Monitor

On each of these non-safety Radiation Monitors, the existing vendor supplied vacuum switch used for high and low flow alarms and pump control interlocks, is deleted. The existing vacuum gauge and in-line rotameter are also deleted. A new in-line mass flow element and a mass flow computer are mounted on the Radiation Monitor skid and connected to provide all of the functions of the vacuum instruments and rotameter.

This modification was precipitated by problems with the vacuum based flow instruments. The failure rate of vacuum flow instrumentation on EMF41 has resulted in a Maintenance Rule classification of A1.

All of the affected radiation monitors are not nuclear safety related. Both the new and the old radiation monitor flow instrumentation are non safety. The instruments installed by this modification perform the same system function as the instruments they replace. Any failure of the new instrumentation will have the identical effect on the operation of their related radiation monitors as failures of the original instruments; however, the new instruments are expected to provide improved accuracy and reliability. This modification assigns Duke instrument mark numbers to the new flow elements and flow computers. These new Duke instruments have been added to the flow diagrams on which their related radiation monitors are shown. Since these flow diagrams are included as figures in the UFSAR, a UFSAR revision will be required to keep the affected figures current.

Evaluation: There are no unreviewed safety questions associated with this modification. This modification has no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required.

11 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-61490 Reroute the discharge piping from Groundwater Drainage Sump Pump C1 and C2

Description: This modification will restore the Groundwater Drainage System (Pump C1) to its intended function of removing groundwater from beneath the floor and the walls of the Auxiliary Building. A leak developed in the discharge piping of Groundwater Drainage System Pump C1 that could not be repaired due to the extensive corrosion of the piping. A reroute of the discharge piping for Groundwater Drainage System Pumps C1 and C2 was required.

The Groundwater Drainage System Pumps C1 and C2 were originally purchased and installed as nuclear safety related equipment. These two pumps were reclassifed as non-nuclear safety related. Seismically designed stainless steel discharge piping was routed from the pump discharge to downstream of the discharge check valves, where it changes to non-nuclear safety related stainless steel piping. Although only a portion of the discharge piping within the sump pump room will be designated as seismically designed, support requirements will provide seismic supports for this piping from pump discharge to the point where the piping exits this room. After the piping exits the sump pump room, it will continue as non nuclear safety related piping up to the point where it enters the Yard. The embedded portion of this section of non nuclear safety related piping is carbon steel. The existing embedded piping, no longer functional, will be cut/capped and abandoned in place.

Evaluation: This modification affects Groundwater Drainage System pumps and piping in the Auxiliary Building and will not degrade the pressure boundary of the remaining embedded piping. There are no Design Basis Events applicable to the location of the new discharge piping, located in the Auxiliary Building. The function or design of the Groundwater Drainage System is not affected by this modification nor does it affect the function or operation of any other system, structure or component. This modification does not involve any unreviewed safety questions. No changes to the Technical Specifications are required. UFSAR text changes are not required. UFSAR Figure 9-195 will be revised to reflect the change of piping classification for Pump C1 and C2 discharge piping.

90 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-61506, Upgrade vestibule S303 (at Aux Bldg Single Point Access) to Fire Protection Related and hold door S303A open, delete the recirculation duct for the vestibule area and delete the power feed associated controls for the fan

Description: A situation has been documented in which the Auxiliary Building Single Point Access (SPA) door, door number S303A was left in the unlatched position. This is a nuclear safety related door and presently serves as a tornado pressure door, an NRC committed fire boundary door, and a ventilation system boundary. The door cannot serve these functions in the unlatched position.

Minor modification CE-61506 will: (1) install a door holder to keep door S303A open during normal situations. The only time it will be closed (by procedure) is upon notification of a tornado warning; (2) upgrade the neighboring vestibule, Room S303, to an NRC committed fire boundary and extend the fire boundary to include the vestibule; (3) upgrade the vestibule area to meet the requirements as a ventilation boundary and extend the ventilation boundary to include the vestibule; (4) delete the Service Building and Warehouse Ventilation System equipment associated with the vestibule; and (5) delete the storage area gate next to the Service Building and Warehouse Ventilation System supply fan.

Adequate design considerations have been made to assure the tornado pressure boundary, the new NRC committed fire boundary and the new ventilation boundary will function properly without introducing any new failures.

Evaluation: There are no unreviewed safety questions associated with modification CE-61506. This change has no effect on the probability or consequences of accidents evaluated in the UFSAR. No Technical Specification changes are needed. The UFSAR will be revised to add a note stating that door S303A will normally be held open, and will only be closed (by procedure) when there is a tornado warning.

204 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-61523, Add sample flow instrumentation to Radiation Monitor 0EMF41

Description: Minor Modification CNCE-61523 changes the sample flow instrumentation on Radiation Monitor 0EMF41, the Auxiliary Building Single Range Beta Monitor.

During Post Mod Testing of Modification CE-61436 it was realized that the vendor's original arrangement of flow instrumentation for 0EMF41 did not perform all of the desired functions. Specifically, the low flow signal that is provided for protection of the sample pump is the same signal that provides the low sample flow alarm. This flow signal, however, is based on total flow through the pump, not the sample flow. The sample pump draws a nominal 12 scfm through twelve sample lines. A network of solenoid valves cycle every 75 seconds to select one sample line at a time for routing through the radiation detector assembly, the remaining eleven sample lines bypass the detector. The bypass line and detector sample line recombine upstream of the sample pump. Given this arrangement, a flow blockage existing anywhere in the sample line selected for monitoring, through the detector assembly to the point at which the sample and bypass lines recombine, will not register as a loss of flow. The loss of 1 scfm through the monitor is merely made up by increased flow through the eleven bypassing lines.

Modification CE-61523 revises the flow instrumentation and controls as follows: The application of 0EMFT5230 is changed to directly trip the sample pump on low flow without imposing a time delay. A new flow element (0EMFE5300) and flow computer (0EMFE5300) are installed to monitor the nominal 1 scfm sample flow that is routed through the detector assembly. The new sample flow monitor provides the loss of sample flow alarm after an appropriate time delay to allow for switching sample lines. Since the total flow instrument trips the pump on abnormal flow, the new flow instrument will respond to the resulting low flow through the detector and will initiate a flow alarm.

Evaluation: EMF41 is not nuclear safety related. Both the new flow detector and computer are non-safety. The instruments installed by this modification provide a function previously thought to exist for 0EMF41. Post modification testting for CE-61436 revealed a deficiency in the existing instrumentation. The new flow instruments are susceptible to a loss of power, but are powered from the same source as EMF41. A loss of power that would disable these instruments would also disable EMF41. Blown fuses in the EMF41 control circuits could disable the flow instruments, but failure of these fuses will also result in a low flow alarm. Any other failure in the new instrumentation will have essentially the same effects on the operation of the EMF as failures of the original instruments; however, the new instruments provide improved flow annunciation by monitoring the actual flow through the detector.

This modification assigns Duke instrument mark numbers to the new flow instruments. These new Duke instruments have been added to the affected flow diagram for 0EMF41. Since flow diagram is included as a figure in the UFSAR, a UFSAR revision will be required to keep the affected figure current. There are no unreviewed safety questions associated with this modification, No Technical Specification changes are required. UFSAR flow diagrams will be revised.

39 **Type:** Minor Modification **Unit:** 2

Title: Minor Modification CE-61528, Remove Power from Main Feedwater Containment Isolation Valve Actuator Nitrogen Solenoid Valves

Description: Minor Modification CE-61528 will attempt to minimize nitrogen leakage problems associated with the Unit 2 Feedwater System Isolation valves (2CF-33, 2CF-42, 2CF-51, and 2CF-60). These valves are 18 inch Borg-Warner pneumatic hydraulic operated gate valves. This modification will remove electrical power from the nitrogen solenoid valves associated with the Main Feedwater Isolation Valves. Removal of power will eliminate cycling of the solenoid valves which can cause seal wear/leakage. This modification will also attempt to prevent nitrogen leakage from solenoid valves which have not yet leaked. The means employed for removal of power is acceptable as a permanent means. Thus, power will not be inadvertently restored due to an unreliable method of isolation. De-energizing these solenoid valves is functionally equivalent to removing them.

The solenoid valves are identified below:

2CFSV0330 Train A and 2CFSV0331 Train B (Steam Generator A, Valve 2CF-33)
2CFSV0420 Train A and 2CFSV0421 Train B (Steam Generator B, Valve 2CF-42)
2CFSV0510 Train A and 2CFSV0511 Train B (Steam Generator C, Valve 2CF-51)
2CFSV0600 Train A and 2CFSV0601 Train B (Steam Generator D, Valve 2CF-60)

Additionally, a replacement nitrogen solenoid valve will be installed for the 2CF-60 valve actuator. The solenoid will not be energized. The replacement is functionally equivalent to the de-energized solenoid valves, and meets the same design requirements as the solenoid valve (2CFSV0601) it is replacing. It will act as a safety related pressure boundary and has been qualified by the vendor. Testing will verify that no degrading effects have been imposed on the main feedwater isolation valve's performance. The vendor drawing and instruction manual will be modified to facilitate a similar replacement for any other main feedwater isolation valve nitrogen solenoid valve.

Evaluation: Consideration was given to making the valve actuator more reliable through less leakage (pneumatic solenoid valve de-energization). These changes have been accomplished while maintaining a safety related, single failure proof design which can isolate the main feedwater isolation valves following a safety related closure signal assuming the single failure of any component. A failure modes and effects discussion justified pneumatic solenoid valve de-energization . The pneumatic solenoid valves are affected by removing power but they are being put in their safe position (open). This modification actually removes a potential failure (opening of the solenoid valves on demand). In summary, no new failure modes are created. A potential failure has been eliminated. No common mode failures are introduced. All applicable design criteria have been preserved in this design.

No unreviewed safety questions are created by this modification. No Technical Specification changes are required. No changes to the UFSAR are required.

40 Type: Minor Modification **Unit:** 1

Title: Minor Modification CE-61529, Remove Power from Main Feedwater Containment Isolation Valve Actuator Nitrogen Solenoid Valves

Description: Minor Modification CE-61529 will attempt to minimize nitrogen leakage problems associated with the Unit 1 Feedwater System Isolation valves (1CF-33, 1CF-42, 1CF-51, and 1CF-60). These valves are 18 inch Borg-Warner pneumatic hydraulic operated gate valves. This modification will remove electrical power from the nitrogen solenoid valves associated with the Main Feedwater Isolation Valves. Removal of power will eliminate cycling of the solenoid valves which can cause seal wear/leakage. This modification will also attempt to prevent nitrogen leakage from solenoid valves which have not yet leaked. The means employed for removal of power is acceptable as a permanent means. Thus, power will not be inadvertently restored due to an unreliable method of isolation. De-energizing these solenoid valves is functionally equivalent to removing them.

The solenoid valves are identified below:

1CFSV0330 Train A and 1CFSV0331 Train B (Steam Generator A, Valve 1CF-33)
1CFSV0420 Train A and 1CFSV0421 Train B (Steam Generator B, Valve 1CF-42)
1CFSV0510 Train A and 1CFSV0511 Train B (Steam Generator C, Valve 1CF-51)
1CFSV0600 Train A and 1CFSV0601 Train B (Steam Generator D, Valve 1CF-60)

Evaluation: Consideration was given to making the valve actuator more reliable through less leakage (pneumatic solenoid valve de-energization). These changes have been accomplished while maintaining a safety related, single failure proof design which can isolate the main feedwater isolation valves following a safety related closure signal assuming the single failure of any component. A failure modes and effects discussion justified pneumatic solenoid valve de-energization . The pneumatic solenoid valves are affected by removing power but they are being put in their safe position (open). This modification actually removes a potential failure (opening of the solenoid valves on demand). In summary, no new failure modes are created. A potential failure has been eliminated. No common mode failures are introduced. All applicable design criteria have been preserved in this design.

No unreviewed safety questions are created by this modification. No Technical Specification changes are required. No changes to the UFSAR are required.

86 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-61531, Restore Valve 1CA-6 to operable status

Description: Modification CE-61531 reverses the changes to valve 1CA-6 made by modification CE-61396 which changed the position of valve 1CA-6 so that the valve is normally closed with the associated motor operator breaker open. After implementation of CE-61531, valve 1CA-6 will be normally open

The changes to the Auxiliary Feedwater System and the Condensate Storage System by modification CE-61531 were needed due to problems with the Auxiliary Feedwater System Condensate Storage Tank (CACST) supply. Currently, valve 1CA-6 is closed with power removed to isolate the CACST from the Auxiliary Feedwater Pump supply piping. This valve is closed to prevent the introduction of air into the Auxiliary Feedwater Pump suction piping. Due to the piping configuration of the Auxiliary Feedwater System and the Condensate Storage System, during certain plant conditions and upon depletion of the CACST, air could be drawn into the Auxiliary Feedwater System suction piping which could result in Auxiliary Feedwater Pump damage. Modification CE-61531 and modification CN-11401/00 resolve the air entrainment issue and restore the CACST to service. Modification CN-50477 will add a dedicated Unit 2 CACST.

Modification CN-11401/00 reroutes and enlarges the Upper Surge Tank (UST) to CACST junction piping and resolves the concern for air entrainment as described in the following paragraph. Modification CE-61531 can not be implemented until modification CN-11401/00 is completed.

The air entrainment issue resolved by modification CN-11401/00 is as follows: Vortex formation in the CACST and USTs could lead to the introduction of air into the suction piping of the Auxiliary Feedwater Pumps, potentially disabling the Pumps. An Operability Evaluation was performed, with support from the Auxiliary Feedwater Pump vendor, which concluded that vortex formation is not an operability concern. During the process of evaluating the vortex concern, a separate mechanism was identified by which air could potentially enter the Auxiliary Feedwater suction piping. This mechanism involved the depletion of the CACST and the failure of valve 1CA-6 (CACST to CA Pump Isolation Valve) to automatically close on a low CACST level. In this situation, the USTs would supply the Auxiliary Feedwater Pumps. However, if condenser vacuum is not broken, the relative elevation head of the UST's with respect to the junction of the Auxiliary Feedwater supply piping from the CACST and USTs is not sufficient to maintain the pressure at this junction above atmospheric pressure over the full range of possible Auxiliary Feedwater flow rates. Also, due to the elevation difference between this junction and the pressure switches that activate the automatic swapover to the assured Auxiliary Feedwater suction sources of the Nuclear Service Water System and the setpoint of these pressure switches, the swapover is not assured if the Auxiliary Feedwater CST/USTs junction pressure is less then atmospheric over the full range of possible Auxiliary Feedwater System flows. This could lead to the introduction of air into the Auxiliary Feedwater suction piping and possibly into the Auxiliary Feedwater Pumps from the depleted and unisolated CACST, potentially disabling the pumps.

Modification CE-61531 does not add or delete any automatic or manual safety related

feature of the Auxiliary Feedwater System, nor does it convert an automatic safety related feature to manual or vice versa. The modification does not introduce an unwanted or previously unreviewed system interaction, but instead eliminates such an interaction. This modification does not alter the QA condition, or seismic or environmental qualification of any component in the Auxiliary Feedwater System as the CACST is a non-safety, non-seismic tank. No adverse effects on the safety related function of the Auxiliary Feedwater System or any interfacing systems are created by this activity. The Condensate Storage System will continue to provide the Technical Specification required condensate inventory of approximately 225,000 gallons. The Nuclear Service Water System will continue to be the nuclear safety related supply for the Auxiliary Feedwater System and this function is not affected. No new failure modes are created by this modification.

Evaluation: No unreviewed safety questions are created as a result of modification CE-61531 which unisolates valve 1CA-6 and thus restores the functionality of the CACST. No Technical Specification changess are required; however, Technical Specification 3.7.6 bases will be revised to reflect the restored CACST. This modification does result in the plant configuration being different from that described in the UFSAR Section 10.4.9.2 . The affected sections and figures will be revised to show the new configuration of the Auxiliary Feedwater System and Condensate Storage System.

211 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-61543, Abandon Level Switch 1NVLS6020 and Associated Annunciation

Description: The Reciprocating Charging Pump has previously been "Removed from Service" by Minor Modification CE-9660. The pump was prevented from operating to prevent violating Chapter 15 Accident Analysis assumptions.

This modification will abandon some of the auxiliary equipment associated with the pump. Specifically, this modification will abandon the level switch (1NVLS6020) for the stuffing box head tank associated with the Reciprocating Charging Pump, and remove the associated Main Control Room annunciation from lamp box location 1AD07.04.04 ("PD PUMP STUFF BOX LO LVL"), which is activated by the level switch. Since the level switch is shown on UFSAR Figure 9-91, the figure will be revised to show the level switch abandoned in place.

The modification will be implemented by lifting the leads to the level switch, which provides the input to annunciator window, 1AD07.04.04 ("PD PUMP STUFF BOX LO LVL"), and replacing the currently engraved annunciator with a blank window.

Evaluation: Minor Modification CE-61543 does not involve an Unreviewed Safety Question. None of the text version of the UFSAR specifically discuss the stuffing box head tank level switch or annunciation provided on low level; therefore, no changes are required to the text of the UFSAR. No Technical Specifications changes are required as a result of this modification. A UFSAR change is required to Figure 9-91 to show the abandonment of level switch NVLS6020.

38 Type: Minor Modification **Unit:** 1

Title: Minor Modification CE-61546 Add Lifting devices in the Annulus for flanges for Containment Vessel Penetrations M-371 and M-394

Description: A personnel safety concern was identified associated with the removal and reinstallation of spare containment penetration flanges, M394 and M371. These penetrations are for access of temporary electrical power, compressed air supply and ice condenser hoses during refueling outages. Modification CE-61546 will provide a lifting device (davit assembly), attached to each penetration sleeve, on the Annulus side, to allow safe removal and replacement of the penetration blind flange.

Penetrations M371 and M394 are in the cylinder wall portion of the Containment Vessel. This steel cylinder supports the Steel Containment Dome and functions as a pressure boundary for the primary containment. The Containment Vessel is designed to limit the release of energy and radioactivity in the event of a Design Basis Event. The davit assemblies will be attached to the existing sleeves for these 20" penetrations, on the Annulus side only. They will be used to manipulate the blind flange on the end of the sleeve when the flange is removed or installed.

These two penetrations and the Containment Vessel are nuclear safety related items. The attachment of the davit assemblies to the penetration sleeves is also nuclear safety related. The remainder of the davit assemblies are not nuclear safety related but are seismically designed.

The davit assemblies for these penetrations will be identical to two similar penetrations, M234 and M452, which have already been implemented for both units.

Evaluation: There are no unreviewed safety questions associated with this minor modification. The containment penetration davit arm assembly is only a lifting device used to assist workers in removal or reinstallation of the penetration blind flanges during refueling outages when it is opened for maintenance activities. The weld of the davit assembly to the sleeve will not degrade the sleeve. Also, the applied torsional moment to the sleeve when the davit is loaded will not degrade the sleeve. The weld to the sleeve will be leak rate tested after the mod is complete to confirm the containment pressure boundary is not degraded. This modification will not affect the operation of the containment penetration. This modification will have no effect on the probability or consequences of accidents evaluated in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required

21 Type: Minor Modification **Unit:** 1

Title: Minor Modification CE-61556, "Change Emergency Lighting Panel Labels (1ELB1 and 2ELB1) to reflect proper panel identification

Description: Minor Modification CE-61556 changes the Emergency Lighting panel labels (1ELB1 and 2ELB1) to reflect proper panel identification. The Emergency Lightning Panels (1ELB1 and 2ELB1) are currently labeled incorrectly. They should be labeled 1ELB and 2ELB.

Evaluation: The labeling of these panels is a non-nuclear safety related change. This activity does not involve any nuclear safety related systems. This change will have no effect on the probability or consequences of accidents analyzed in the UFSAR. There are no unreviewed safety questions associated with this change. No Technical Specification changes are required. Selected Licensee Commitments Manual Tables 16.8-1A and 16.8-1B will be revised.

4 Type: Minor Modification **Unit:** 1

Title: Minor Modification CE-61561, Abandonment of 1CMFT5932 and 1CMFT5942 and removal of associated equipment

Description: Flow transmitters 1CMFT5932 and 1CMFT5942 were originally installed to monitor the feedwater pump suction flow and provide an input to the Feedwater Control System). The signals from the transmitters resulted in a flow error signal for the feedwater pump speed control system, which was used to provide automatic flow balancing of the Feedwater System. However, operational problems with the system design at McGuire Nuclear Station resulted in the removal of the transmitter signal from the Feedwater Control System at Catawba prior to start-up via modification CN-10227.

A Station Problem Report was initiated in 1989 to document that the signals from 1CMFT5932 and 1CMFT5942 were deleted from the Feedwater Control System via modification CN-10227, but design documentation did not properly reflect as-built conditions. A Variation Notice associated with minor modification CE-2277 updated the Instrumentation and Control list by adding a note stating that the transmitters were no longer used and revised the Instrument Detail drawing to remove the transmitters functional description. However, all associated design documentation still did not accurately reflect that the transmitters were not used, which resulted in the initiation of the plant problem report. Therefore, this modification was initiated to document the abandonment of the transmitters, and remove unnecessary associated components.

The instruments removed are the Square Root Extractors (Converters) and the Signal (Current-to-Voltage) Converters associated with the abandoned flow transmitters.

Evaluation: There are no unreviewed safety questions associated with this modification. The modification will allow the as built condition of the plant to be correctly shown on plant drawings. The modification will have no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required.

15 Type: Minor Modification **Unit:** 0

Title: Minor Modification CE-61564, Remove the Kirk Key Interlock scheme associated with the 230 KV Switchyard 125 VDC Power System battery breakers and distribution center cross tie connection

Description: Minor Modification CE-61564 deletes the Kirk Key Interlock scheme associated with 230KV Switchyard 125VDC battery breakers and the distribution center cross tie connection. Per the Switchyard Systems Design Basis Specification, to equalize one of the batteries, the first step is to connect the two distribution centers together and then disconnect the battery to be equalized. The order is to assure that DC power is available should the normal charger fail or a blackout occur. The preferred breaker operation sequence cannot be accomplished with the Kirk Key scheme as it is currently installed. One of the battery breakers must be opened first to release its Kirk Key, which must then be inserted in the cross tie breaker Kirk Key cylinder, permitting breaker closure.

The installed Kirk Key Interlock scheme involves Switchyard 125VDC Power System distribution center circuit breakers installed in SYD-1 Compartments F01A, F02C and SYD-2 Compartment F01A. The interlock scheme is comprised of three Kirk Key assemblies, each with identically keyed cylinders, but only two Kirk Keys available. The three breaker compartments are interlocked such that only two of the three breakers may be closed at the same time. The 230kV Switchyard 125VDC System battery breaker/distribution center cross tie connection Kirk Key Interlock scheme is not described in the UFSAR, Catawba Selected Licensee Commitments or Technical Specifications. SER, Section 8.2.1, refers generally to interlock schemes associated with the Switchyard 125VDC Power System: "The 125VDC supply for the 230kV switchyard relays, control, and monitoring equipment consists of two separate supplies, each with its own battery and battery charger, with a spare battery charger interlocked to serve either system separately. The first supply serves a train A primary relaying and control circuit as well as a train B secondary relaying and control circuit. The second supply serves a train B primary relaying and control circuit as well as a train A secondary relaying and control circuit. The two supplies are tied together in an interlocking scheme that allows manual power transfer but no more that one source of power to either bus. The system has adequate protection, monitoring and alarm in the switchyard relay house and in the Catawba Control Room."

The Switchyard Systems Design Basis Specification describes the preferred steps to establish the system line-up necessary for applying an equalize charge to either battery, using the shared battery charger: " . . the first step is to connect the two distribution centers together and then disconnect the battery to be equalized." The purpose of connecting the two distribution centers is to assure DC power will be available, from the opposite distribution center's battery, should either of the normal chargers fail or lose its power supply (black out). The original system description stated: "If either battery is lost or has to be removed for servicing, the distribution centers are to be tied together (cross connected) through the bus tie breakers." Conversely, if both batteries are in service and are connected to their respective charger and distribution center, the distribution centers would not be cross-connected. When cross-connected, both distribution centers are susceptible to the same fault conditions that would otherwise affect only one distribution center, if not cross connected.

Operating a DC bus, powered only by a charger without a floating battery has been found to be undesirable in LOOP event at another Duke Power Site. Other Catawba battery systems procedures include a warning to prevent operating a charger without a battery connected.

Minor Modification CE-61564 deletes the Kirk Key Interlock scheme associated with the system Switchyard 125VDC Power System battery breakers and distribution center cross tie connection. The minor modification will revise applicable Duke and Vendor documents and identify IP/0/B/3710/030, Switchyard 125VDC Power System Battery Equalize Procedure, as requiring revision. Breaker operational control, affected by the deletion of the "mechanical' interlock scheme, will be controlled via the Switchyard 125VDC Power System Battery Equalize Procedure.

Evaluation: Deleting the Kirk Key Interlock scheme (a manual feature), associated with the Switchyard 125VDC Power System battery breakers and the distribution center cross tie connection, will have no effect on current system design for normal operation. The distribution center cross tie connection is utilized when either battery, SYB-1 or SYB-2, is to be removed for equalization by spare charger SYBC-S. The probability of a fault, which could affect both distribution centers, if cross connected, is small. Operating a DC system, without a battery connected, was identified as a major contributor to locking out both the Red and Yellow buses at another nuclear station in a 1992 industry event. Without a battery connected, high charger output voltage caused a series of spurious relay actuations. Compared to a complete loss of switchyard, the risk of a fault, while the distribution centers are cross connected, is minimal and deemed acceptable.

Deletion of the Kirk Key Interlock scheme will not increase the probability or consequences of accidents previously evaluated in the UFSAR. The switchyard 125VDC Auxiliary Power System is not identified as a cause or contributor in any Catawba accident analysis.

There are no unreviewed safety questions associated with this minor modification. No Technical Specification changes are required. No UFSAR changes are required

93 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-61571, Retag two Nuclear Sampling System grab sample valves

Description: This modification will replace "descriptive name" tags on two Nuclear Sampling System grab sample valves with Nuclear Sampling System "valve number" tags. The two Nuclear Sampling System grab sample valves to be re-tagged are located in Nuclear Sampling System Sample Sink 1B (located in the Auxiliary Building at elevation 543', Room 238) and Nuclear Sampling System Sample Sink 2A (located in the Auxiliary Building at elevation 543', Room 248). The valve tag numbers that will be used for the grab sample valves are 1NM938 and 2NM938. This modification affects non nuclear safety related piping and equipment located in the Auxiliary Building.

The grab sample valves to be re-tagged provide a convenient, shielded and safe means for obtaining a sample from the Nuclear Sampling System Automation Sample Panel. The Nuclear Sampling System Automation Sample Panel is comprised of both non nuclear safety related piping and radwaste related piping and tubing. This modification affects radwaste related piping portion of Nuclear Sampling System tubing located in the Auxiliary Building, and does not degrade the pressure boundary of this tubing. There are no Design Basis Events or Design Events applicable to the re-tagging of the two grab sample valves.

Since these valves (Whitey part # SS-3NKRS4-P) are 2500 lb. ANSI Class 316ss, they are suitable for this application. Neither the function nor design of the Nuclear Sampling System is affected by this modification, nor does it affect the function or operation of any other system, structure or component. The valve tags added by this modification will comply with the requirements for process valves in accordance with Catawba Site Directive 3.0.5.

Evaluation: This modification does not involve any unreviewed safety questions. No changes to the Technical Specifications are required. UFSAR text changes are not required, however UFSAR Figure 9-81 will be revised to reflect the routing of the grab sample valve (Unit 1) from the Nuclear Sampling System Automation Sample Panel to Sample Sink 1B.

249 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-61575, Construct a new chlorine storage facility that will allow chlorine to be stored further away from the Control Room Air Intakes

Description: Minor Modification CE-61575 will relocate the existing chlorine gas storage facility to a new building that will be placed in the storage yard north of the motor pool garage. The new storage location is approximately 2200 feet from the nearest control room air intake. The current chlorine gas storage area located in the bottle gas storage house will be abandoned. It is approximately 415 feet from the nearest control room air intake. The UFSAR will be revised to describe the new chlorine gas storage location. This activity is being performed to support Nuclear Station Modification CN-50486, which will replace the current safety related chlorine detectors, (which sample from the control room air intake ductwork) with non-safety related detectors.

Evaluation: The reason for relocating the chlorine storage facility is to provide additional assurance that an accidental chlorine spill will not impact control room habitability. The current storage facility is approximately 415 ft. away from (and almost in direct line with) the Unit 1 control room air intake. By increasing the distance and eliminating any direct flowpaths, the impact of any potential chlorine spill will be greatly diminished. The new storage location will contain the same type of chlorine storage containers (150 pound bottles) as the original location, and approximately the same quantity of chlorine. Because the distance from the control room intakes has increased, but all other storage requirements remain the same, the effect of a postulated chlorine leak will be reduced due to greater dilution and dispersal of the chlorine cloud as it travels a longer distance to the intake. The current storage location will be abandoned, and no chlorine will be stored in that area. The proposed change is acceptable because it does not increase the consequence of a postulated chlorine leak.

The UFSAR lists the chlorine storage locations in Section 2.2.3.1.4. The UFSAR revision will delete the Gas Bottle Storage House from the listing, and replace it with the Chlorine Storage Facility. This is the only place in the UFSAR where this chlorine storage location is listed.

139 Type: Minor Modification **Unit:** 1

Title: Minor Modification CE-61590, Abandon Steam Generator Blowdown Tank Vent Flow Line

Description: The Steam Generator Blowdown System is used in conjunction with the Condensate System to maintain proper secondary side water chemistry. Non-volatile solids resulting from corrosion, steam generator tube leaks, or condenser tube leaks tend to concentrate in the steam generators. The Steam Generator Blowdown System is designed to control the concentration of these impurities by continuously removing a portion of fluid from the shell side of the steam generators. The blowdown is either discarded or purified for makeup to the Condensate Makeup System.

The Steam Generator Blowdown System begins at the steam generator blowdown nozzles where blowdown flow is extracted from the steam generators. The blowdown flow is routed through flow control valves to the blowdown tank. The blowdown tank separates the water and steam phases of the blowdown. The steam phase is normally routed to the "D" heater extraction lines to recover thermal energy and conserve condensate. Alternatively, the steam from the blowdown tank can be vented to atmosphere.

The blowdown tank pressure is controlled to a constant value by valve 1BB-250, which is located in the vent line to the "D" heater. Flow instrumentation was originally installed upstream of valve 1BB-250 to monitor the flow rate of the steam leaving the tank. The instrumentation consists of a flow element (1BBFE5250), pressure transmitter (1BBPT5260) and a flow computer (1BBP5250). In addition, the flow computer also provides a signal to the Operator Aid Computer.

A request that the flow instrumentation be abandoned was initiated due to the fact that it did not function and was not needed for plant operation. This modification will abandon flow element 1BBFE5250 and remove flow computer 1BBP5250 and pressure transmitter 1BBPT5260. The modification will also disconnect the output of the flow computer from the Operator Aid Computer and void all circuits associated with the instrumentation.

Evaluation: UFSAR Section 10.4.8.5.1 states that: "Flow instrumentation is provided in the Blowdown Tank steam vent line to indicate steam flowrate leaving the tank. "

The Steam Generator Blowdown Tank vent line flow instrumentation is Non-Safety Related. The flow instrumentation is currently not functioning and is it required for any thermal power or thermal performance calculations. In addition, the flow instrumentation is not used to control any other plant components. Therefore, the abandonment of the flow element 1BBFE5250, the removal of flow computer 1BBP5250 and pressure transmitter 1BBPT5260, and the removal of associated OAC Computer Point C1A1423 will not adversely impact plant operation.

There are no unreviewed safety questions associated with this UFSAR Change. No Technical Specification changes are required. UFSAR Section 10.4.8.5.1 will be revised.

140 **Type:** Minor Modification **Unit:** 2

Title: Minor Modification CE-61591, Abandon Steam Generator Blowdown Tank Vent Flow Line

Description: The Steam Generator Blowdown System is used in conjunction with the Condensate System to maintain proper secondary side water chemistry. Non-volatile solids resulting from corrosion, steam generator tube leaks, or condenser tube leaks tend to concentrate in the steam generators. The Steam Generator Blowdown System is designed to control the concentration of these impurities by continuously removing a portion of fluid from the shell side of the steam generators. The blowdown is either discarded or purified for makeup to the Condensate Makeup System.

The Steam Generator Blowdown System begins at the steam generator blowdown nozzles where blowdown flow is extracted from the steam generators. The blowdown flow is routed through flow control valves to the blowdown tank. The blowdown tank separates the water and steam phases of the blowdown. The steam phase is normally routed to the "D" heater extraction lines to recover thermal energy and conserve condensate. Alternatively, the steam from the blowdown tank can be vented to atmosphere.

The blowdown tank pressure is controlled to a constant value by valve 2BB-250, which is located in the vent line to the "D" heater. Flow instrumentation was originally installed upstream of 2BB-250 to monitor the flow rate of the steam leaving the tank. The instrumentation consists of a flow element (2BBFE5250), pressure transmitter (2BBPT5260) and a flow computer (2BBP5250). In addition, the flow computer also provides a signal to the Operator Aid Computer.

It was requested that the flow instrumentation be abandoned due to the fact that it did not function and was not needed for plant operation. Therefore, this modification will abandon flow element 2BBFE5250 and remove flow computer 2BBP5250 and pressure transmitter 2BBPT5260. The modification will also disconnect the output of the flow computer from the Operation Aid Computer and "Void" all circuits associated with the instrumentation.

UFSAR Section 10.4.8.5.1 states "Flow instrumentation is provided in the Blowdown Tank steam vent line to indicate steam flowrate leaving the tank."

Evaluation: The Steam Generator Blowdown Tank vent line flow instrumentation is not Nuclear Safety Related. The flow instrumentation is currently not functioning nor is it required for any thermal power or thermal performance calculations. In addition, the flow instrumentation is not used to control any other plant components. Therefore, the abandonment of the flow element 2BBFE5250, the removal of flow computer 2BBP5250 and pressure transmitter 2BBPT5260, and the sparing of the associated Operator Aid Computer Point will not adversely impact plant operation.

There are no unreviewed safety questions associated with this UFSAR Change. No Technical Specification chnages are required. UFSAR Section 10.4.8.5.1 and USFAR Figure 10-29 will be revised.

159 Type: Minor Modification **Unit:** 0

Title: Minor Modification CE-61602 - Abandon Waste Gas System Level Switches

Description: The Waste Gas System Low Point Drain Monitoring System was installed by modification CN-50110. The system was installed to alleviate the plugging of the Waste Gas System drain header and allow the drain header to function more efficiently. However, due to the constant problems (nuisance alarms) with the system, a station problem report requested that the instrumentation be abandoned due to the fact that it did not function properly and maintenance of the equipment was difficult because the sensors associated with the level switches were located in a high radiation area. Radwaste Chemistry and System Engineering developed a method to operate the Waste Gas System without reliance on this monitoring system. Therefore, this modification was initiated to abandon this level instrumentation system.

The abandonment of the level monitoring system will result in the reduction of cost as well as provide an ALARA cost savings due to not having to maintain the instrumentation.

This modification will affect the Waste Gas System Low Point Drain Monitoring System by abandoning level switches 0WGLS6180, 0WGLS6190, 0WGLS6200, 0WGLS6210, 0WGLS6220, 0WGLS6230, 0WGLS6240, 0WGLS6250 and 0WGLS6260. The associated reflash module and annunciator window on Waste Gas Panel will also be spared. In addition, two circuits associated with the instrumentation will be voided.

Evaluation: The Waste Gas System Low Point Drain Monitoring System is not nuclear safety related. The level instrumentation is currently not functioning properly. In addition, the level instrumentation is not used to control any other plant components. Therefore, the abandonment of level switches 0WGLS6180, 0WGLS6190, 0WGLS6200, 0WGLS6210, 0WGLS6220, 0WGLS6230, 0WGLS6240, 0WGLS6250 and 0WGLS6260, and the sparing of the reflash module and annunciator window on Waste Gas Panel will not adversely affect plant operation.

There are no unreviewed safety questions associated with this modification. Since UFSAR Figure 11-28 shows the level instrumentation, changes to the UFSAR are required for this modification. However, since the instrumentation is not providing any signals to any plant control systems, the abandonment of the instrumentation will not adversely affect the operation of the plant during Normal, Abnormal or Accident conditions. No Technical Specification changes are required.

161 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-61625, Replace the Existing 1A Chemical and Volume Control System Centrifugal Charging Pump Seals with New Third Generation Seals and Add a Pressure Gauge to Monitor and Trend the Pump Balance Drum Pressure

Description: This modification replaces the existing second generation Chemical and Volume Control System 1A Centrifugal Charging Pump mechanical seals with third generation seals supplied by the pump OEM, Ingersoll Dresser Pumps (IDP). The current inboard and outboard seals used on the pumps do not provide adequate service life, involve a significant risk of unrepairable shaft damage during maintenance, and are not the configurations on which active design improvements are made. IDP has shown this seal upgrade and conversion as suitable and appropriate for use with the Chemical and Volume Control System Emergency Core Cooling System (ECCS) pumps.

The third generation conversion seals do not require the flush cooling system that is currently used on the second generation seals. The leak prone tubing that supplies the flush water to the seals from the seal flush adapters will be removed and three of the four the tubing connections on the flush adapters will be plugged. A pressure gauge that will be used for trending of the performance of the pump balance drum will be installed in the upstream side connection on the outboard (thrust end) balance line flush adapter. The new gauge will be read on operator tours to trend pump performance.

The pump shaft vibration probe brackets that are attached to each seal housing face will be modified from their current design to also function as splash shields. This will be accomplished by fabricating two new brackets that will extend further around the pump shaft. IDP has agreed to drill and tap the new seal housings for the third generation seals that will be supplied to accommodate the attachment and mounting of the probe brackets.

Evaluation: The new seals have been supplied by IDP to meet all applicable UFSAR Sections and ASME Section III, Class 2 codes to operate as acceptable safety related replacement seals for the existing 1A Chemical and Volume Control System Centrifugal Charging Pump. The probability of occurrence of an accident or malfunction as previously evaluated in the SAR is not increased. The seals have been certified by IDP to be equivalent to the second generation seals as to form, fit, and function, and do not degrade the ability of the Chemical and Volume Control System to perform as designed to mitigate an accident. The use of the new third generation seals have been evaluated and have been determined to be acceptable for use on the 1A Centrifugal Charging Pump.

The Chemical and Volume Control System Centrifugal Charging Pump seal replacement does not change the operation or design basis of the Chemical and Volume Control System or any other system that is described in the SAR. This modification does not add any new failure modes or operating characteristics. The failure mode described in the SAR is for the Chemical and Volume Control System 1A pump to fail to deliver the working fluid at the prevailing Reactor Coolant System pressure. The failure would reduce the redundancy of providing charging and seal water flow to the Reactor Coolant System . If this were to occur, the alternate 1B Centrifugal Charging Pump would provide the minimum flow delivery requirements at prevailing high Reactor Coolant System pressure. There are no new failure modes identified as a result of this modification. Small pipe breaks, such as tubing rupture, have been previously evaluated in the SAR. Margins

associated with the integrity of the fission products barriers are not exceeded by this activity. There are no changes to any setpoint, design limit, or operating parameter. The new seals will not cause the 1A Centrifugal Charging Pump to operate outside its design basis and pose no change in the method of normal operation and emergency operation.

The scope of this modification does not involve an unreviewed safety question. No Technical Specification changes are required. UFSAR Figure 9-96 will be revised to show the flow balance line and pressure gauge (1NVPG7200) on the 1A Centrifugal Charging Pump.

137 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-61629 Provide two penetrations in the Equipment Hatch Seal Ring to allow access to the annulus

Description: Minor Modification CE-61629 adds two nuclear safety related penetrations in the Unit 1 Concrete Containment Equipment Hatch penetration seal ring sleeve to allow access to the annulus. This modification will involve providing two penetrations that are normally sealed by a bolted cover and gasket. The purpose of this modification is to provide another access to the annulus for routing temporary services (such as Steam Generator Maintenance, Ice Condenser Maintenance, Temporary Power, Integrated Leak Rate Testing) hoses and cables during the outages. Currently, the only access to the Annulus from outside the Reactor Building during outages is through the Equipment Hatch Boot Seal. The boot seal causes a restriction on the size of hoses that can be used which is less than three inches. The Integrated Leak Rate Test requires the capacity of multiple diesel air compressors to pressurize containment during this test. Providing the new penetrations will allow larger hoses to be used for the Integrated Leak Rate Test which will result in quicker pressurization of containment. In addition, providing the new penetrations will help prevent future degradation of the boot seal. This degradation occurs when the boot seal is unbolted and moved to allow hoses and cables to be routed to the annulus.

There are three major penetrations through the concrete shell wall and containment cylinder wall. These penetrations are

1. Equipment Hatch
2. Lower Personnel Air Lock
3. Upper Personnel Air Lock

This modification affects the Equipment Hatch penetration. The Equipment Hatch penetration is a 20'-0" inside diameter rolled sleeve. This sleeve is not designed to carry any loads and serves primarily as a concrete form during construction. During normal operation, this large penetration is covered by the Equipment Hatch Gate. A "boot" seal is located between the Concrete Containment Equipment Hatch penetration seal ring sleeve and the "barrel" of the Steel Containment Equipment Hatch penetration sleeve. This seal serves three purposes:

1. to maintain the pressure boundary for the Annulus Ventilation System
2. to maintain the pressure boundary for tornado pressure/depressurization
3. to accommodate the separation (Y gap) between the Concrete Containment and the Steel Containment.

Currently, during outages a section of the boot seal is moved to allow temporary hoses and cables to be routed into the annulus. The size of the boot seal opening restricts the size of the cables and hoses that can be routed through it and moving the boot seal to allow this access contributes to the degradation of the seal. The two new seal ring penetrations provided by this modification will be located in the Concrete Containment Equipment Hatch penetration seal ring sleeve. They will be designated as Equipment Hatch Seal Ring Sleeve Temporary Services Penetration C401 and Equipment Hatch Seal Ring Sleeve Temporary Services Penetration C402. These penetrations may be used to access the annulus instead of using the boot seal.

Calculation CNC-1144.02-04-0002, Rev. 0 has been performed to qualify the design of the two new penetrations. During normal operation, the penetration openings will be sealed by a nuclear safety related cover plate and gasket that is bolted to the Concrete Containment Equipment Hatch penetration seal ring sleeve. A divider bar is welded to the seal ring sleeve in the center of the opening to satisfy the security requirement to maintain the area of the penetrations less than 96 square inches. The cover plate, bolting, divider bar and gasket materials are suitable for the design conditions in the annulus and equipment hatch penetration. The gasket material is the same material used for the boot seal.

Evaluation: Based on calculation CNC-1144.02-04-0002, Rev. 0, the function or integrity of the Concrete Containment Equipment Hatch penetration seal ring sleeve will not be affected by the new penetrations C401 and C402. The penetration design (cover plate, bolting, divider bar and gasket) has been qualified for the applicable design loads (seismic, annulus ventilation system pressure, tornado pressure). The design or function of any other reactor building structure will not be affected.

This modification does not involve an unreviewed safety question. No changes to the Technical Specifications are required. UFSAR Figure 3-264 (Containment Vessel Penetration Details), Figure 3-278 (Reactor Building. Pressure Seals and Gaskets) and Table 6-77 (Containment Isolation Valve Data) will be revised to reflect the addition of the new penetrations added by this modification.

146 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-61636, Remove Minor Modifcation CE-61633

Description: Minor Modification CE-61636 will restore the Nuclear Service Water System Pit A level instrumentation to its original configuration, which is used to protect the plant against a loss of Lake Wylie event, by undoing the changes made to the instrumentation per Minor Modification CE-61633.

Minor Modification CE-61633 did include changes to UFSAR Section 9.2.1.5.5 based on the changes made per the modification. Since the modification is being undone, the proposed UFSAR changes will need to be deleted from the UFSAR.

Evaluation: There is no Unreviewed Safety Question associated with this UFSAR change. This change returns the plant to its original design. This UFSAR Change has no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. UFSAR Section 9.2.1.5.5 will be returned to its original text.

162 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-61659, Delete Boric Acid Tank Diaphragm

Description: Minor Modification CE-61659 will delete the Boric Acid Tank diaphragm completely. The diaphragm will be cut out just above the mounting ring. There will be no need to remove the nuts and bolts that hold it in place. The instrument for reading pressure under the diaphragm (1NVPG6070) will be tagged removed from service.

The Boric Acid Tank is a safety related tank used solely as a backup means for ensuring an adequate shut down margin. The original purpose of the Boric Acid Tank diaphragm was to keep dissolved oxygen out of the Boric Acid Tank. The dissolved oxygen source for the Boric Acid Tank is the Boric Acid Batching Tank. The Boric Acid Batching Tank is an open tank wherein boric acid and water are mixed together to produce the boric acid solution used to operate the unit. When the contents of the Boric Acid Batching Tank are transferred to the Boric Acid Tank, all the dissolved oxygen is transferred to the tank also. The conclusion can then be drawn that the diaphragm in the Boric Acid Tank cannot serve any function since the water that is pumped into them from the Boric Acid Batching Tank is already high in dissolved oxygen.

The diaphragm installed in the Unit 1 Boric Acid Storage Tank has passed its expected lifetime. If the diaphragm should degrade and sink, it could potentially stop up the Boric Acid Tank Pump suction thereby eliminating the ability of the operator to add boric acid to the Reactor Coolant System or Refueling Water Storage Tank. The diaphragm does not provide any Boric Acid Tank overflow protection.

In order to maintain the viability of the Boric Acid Tank as a Technical Specification borated water source, the Boric Acid Tank diaphragm must be removed. The diaphragm does not serve a major function in keeping oxygen out of the Boric Acid Tank since boric acid is made in the Boric Acid Batching Tank which is open to atmosphere. Chemistry sampling reveals that the Boric Acid Tank has boric acid that is close to being oxygen saturated.

Experience at other plants has shown that operating without the diaphragm in the Boric Acid Tank does not affect the operation of the unit. It was noted that there was some small increase in dissolved oxygen in the Boric Acid Tank, however it was still within the capability of the Chemical and Volume Control System to deal with it.

Pressure gauge 1NVPG6070 will be removed from service. The instrument serves as an indication of the condition of the diaphragm. It serves no safety function for the Boric Acid Tank (the Boric Acid Tank is vented to atmosphere).

The diaphragm does not affect the safety function of the Boric Acid Tank. The weight of the diaphragm is negligible and will not affect the seismic qualification of the tank. The diaphragm and pressure gauge will have no effect on the safety function provided by tank level instrumentation.

Evaluation: This modification does not affect or degrade the ability of the Chemical and Volume Control System or any other system that interfaces or interacts with the Chemical and Volume Control System Boric Acid Tank to perform its design or safety function to

mitigate accident consequences. There are no unreviewed safety questions associated with removal of the diaphragm. No Technical Specification changes are required. No UFSAR Changes are required.

112 **Type:** Minor Modification **Unit:** 2

Title: Minor Modification CE-61660, Delete the Boric Acid Tank Diaphragm

Description: Minor Modification CE-61660 will remove the Boric Acid Tank diaphragm. The diaphragm will be cut out just above the mounting ring. There is no need to remove the nuts and bolts that hold it in place. The instrument for reading pressure under the diaphragm 2NVPG6070 will also be tagged removed from service.

Evaluation: The Boric Acid Tank (BAT) is a safety related tank used as a backup means for ensuring an adequate shutdown margin. The original purpose of the BAT diaphragm is to keep dissolved oxygen out of the tank. The dissolved oxygen source for the BAT is from the Boric Acid Batching Tank (BABT). The BABT is an open tank where boric acid and water are mixed together to produce the boric acid solution used to operate the unit. When the contents of the BABT are transferred to the BAT, all the dissolved oxygen is transferred to the tank also. It can be concluded that the diaphragm in the BAT does not serve any function since the water that is pumped into the BAT from the BABT is already high in dissolved oxygen.

The diaphragm installed in the Unit 2 Boric Acid Storage Tank has passed its expected lifetime. If the diaphragm should degrade and sink, it could potentially restrict the BAT Pump suction thereby compromising the ability to add boric acid to the Reactor Coolant System or the Refueling Water Storage Tank. The diaphragm does not provide any BAT overflow protection.

In order to maintain the viability of the BAT as a Technical Specification borated water source, the BAT diaphragm will be removed. The diaphragm does not serve a major function in keeping oxygen out of the BAT since boric acid is made in the Boric Acid Batching Tank which is open to atmosphere. Chemistry sampling shows that the BAT has boric acid that is near to being oxygen saturated.

Experience at another Duke Energy nuclear plant has shown that operating without the diaphragm in the BAT does not affect the operation of the unit. This will cause a small increase in dissolved oxygen in the BAT, however it will still be within the Chemical and Volume Control System's capability.

Pressure gauge 2NVPG6070 will be removed from service. The instrument serves as an indication of the condition of the diaphragm. It serves no safety function for the BAT (the BAT is vented to atmosphere).

The diaphragm does not affect the safety function of the BAT. The weight of the diaphragm is negligible and does not affect the seismic qualification of the tank. The diaphragm and pressure gauge will have no effect on the function provided by the tank level instrumentation.

This modification does not affect the ability of the Chemical and Volume Control System or any other system that interfaces or interacts with the BAT to perform its design or safety function to mitigate accidents.

There is no unreviewed safety question associated with this modification. No Technical

Specification changes are required. No UFSAR changes are required.

188 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-61684, Remove Unit 1 Chemical and Volume Control System Mixed Bed Demineralizer Resin Strainer and Insert a Spacer Plate in Place of its Base

Description: Minor modification CE-61684 removes the Unit 1 Chemical and Volume Control System Mixed Bed Demineralizer Resin Strainer and inserts a spacer plate in place of its base. The design of the spacer plate insert was part of the modification.

This strainer was originally installed as a backfit at Catawba due to a valve mispositioning at McGuire Nuclear Station that resulted in Mixed Bed Demineralizer resin being sluiced back into the inlet piping during their startup. Catawba was backfitted with cone strainers installed in reverse to the normal Chemical and Volume Control System letdown flow. It appears that the wire mesh in the cone strainers is not adequate to handle flowrates normally experienced in the letdown line in the reverse direction. Also by the nature of the strainer location, this strainer collects hot particles and becomes highly radioactive.

Due to the dose rates and the parts availability, Catawba Engineering concluded that the strainer should be permanently removed from the system. The inlet valves to the Mixed Bed Demineralizer are procedurally closed whenever a Mixed Bed Demineralizer is sluiced out for reloading. If by chance some resin did get by the closed inlet valve and into the inlet piping, it would be flushed back into the Mixed Bed Demineralizer as soon as normal flow was re-established. Also, the Mixed Bed Demineralizers are normally sluiced out while letdown is in service. Therefore, due to the pressure in the letdown piping, there is no chance that the sluicing of a Mixed Bed Demineralizer would have enough pressure to force resin out through the closed inlet isolation valve into the pressurized letdown piping.

Evaluation: This modification involves no unreviewed safety questions. The change will have no effect on the probability or consequences of accidents analyzed in the UFSAR. No changes to the Technical Specifications are required. UFSAR Figure 9-92 and UFSAR Section 9.3.4.2.3.11 will be revised.

250 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-61687, Provide installation of a flange assembly at valve 1RNE45

Description: Minor Modification CE-61687 concerns installation of a permanent flush/drain connection on Elevation 577' in the Auxiliary Building. This connection will be used during various Nuclear Service Water System flushes. The new connection will consist of a flange assembly at valve 1RNE45 (Nuclear Service Water System Non-Essential Return Header Drain Valve). This Minor Modification will provide the installation of the flange assembly at valve 1RNE45 on the Nuclear Service Water System Unit 1 Non-essential Return Header. An approved station procedure will be required to control any connection to this flange assembly. The safety evaluation for any procedure used to connect to this assembly would be required to address the effect the connection would have on the Nuclear Service Water System and any other system, structure or component.

Evaluation: The Catawba Nuclear Service Water System, including Lake Wylie and the Standby Nuclear Service Water Pond, is the ultimate heat sink for various nuclear safety related heat loads during normal operation and design basis events. The Nuclear Service Water System also supplies emergency makeup water to various nuclear safety related systems during postulated design basis events, water for fire protection hose stations in the diesel buildings and Nuclear Service Water System Pumphouse, and cooling flow and flush water for non-nuclear safety related heat loads and functions during normal operation.

This Minor Modification affects the Nuclear Service Water System Unit 1 Non-essential Return Header on Elevation 577-00 in the Auxiliary Building. The Nuclear Service Water System nonessential headers primarily have a Class G pressure boundary. Nonessential piping is seismically designed where flooding is a concern in the Auxiliary Building, and Class B piping is used for containment penetrations. The nonessential header on each unit provides flow or can provide flow to various non-safety components. This Minor Modification will have no effect on the function or design basis of the Nuclear Service Water System or any other system.

The design requirements of this portion of the Nuclear Service Water System are Duke Class G, carbon steel, 150 psig and 150 degrees F. The flange assembly at valve 1RNE45 is carbon steel and will meet the design temperature and pressure requirements of the Nuclear Service Water System as stated above. Carbon steel is an acceptable material for use in this portion of the Nuclear Service Water System. This flange assembly is suitable for draining and flushing functions. The stress analysis for the affected portion of the Nuclear Service Water System has been evaluated to account for the additional weight of the flange assembly and the existing seismic and support design is adequate.

The flange assembly at valve 1RNE45 will not affect the integrity or function of the Nuclear Service Water System or any other system, structure or component. This flange assembly will be utilized as a connection for draining and/or flushing activities that are controlled by an approved station procedure. No new failure modes or operating characteristics are created. The Nuclear Service Water System will continue to function as designed.

There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. UFSAR Figure 9-28 (a piping flow

drawing) will be revised.

2 **Type:** Minor Modification **Unit:** 2

Title: Minor Modification CE-70074 Hotwell Pump 2A Cartridge Type Mechanical Seal

Description: Modification CE-70074 involves replacement of the Hotwell Pump 2A mechanical seal from a component-type seal to a cartridge-type seal. The replacement seal introduces no new failure modes and is expected to have reliability equal to or greater than the original seals. Installation of the new seals requires some minor seal flush piping changes that require flow diagram changes.

Evaluation: There are no unreviewed safety questions associated with this modification. The modification will have no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required.

5 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-70107, Replace valves 1ZP-30 and 1ZP-36

Description: Minor Modification CE-70107 will replace valves 1ZP-30 and 1ZP-36 (Item Number CMV-662) with new valves (Item Number DSV-031). Presently these valves are one inch T-type globe valves. The replacement valves are one inch ball valves that are suitable for the application. All affected drawings will be revised to reflect this new information.

The Vacuum Priming System is used to continuously remove air from cooling water lines, condenser waterboxes and other heat exchangers that operate under a vacuum on the secondary side of the plant. Valves 1ZP-30 and 1ZP-36 are the 1A2 and 1B1 Main Condenser Waterbox Priming Valve outlets.

Evaluation: The new valves have been evaluated to be a suitable replacement for the existing valves. There are no Unreviewed Safety Questions associated with this modification. Flow diagram CN 1598-03.01 (not a UFSAR Figure) will be revised to show the new valve designs. No Technical Specification changes are required. No UFSAR changes are required.

3 **Type:** Minor Modification **Unit:** 2

Title: Minor Modification CE-70111, Change Hydrogen Igniter ignition tranformer wiring from the 14 Volt Tap to the 12 Volt tap

Description: Minor Modification CE-70111 changes the hydrogen ignitor transformer tap connection from the 14 Volt tap to the 12 Volt tap. Presently, the 14 Volt tap is used which actually applies approximately 15 volts to each of the hydrogen ignitors. There has been an excessive failure rate with 15 volts supplied to the ignitors. At the lower tap location, approximately 13 volts will be applied to the ignitors. Ignitors were bench-tested at 13 volts and their ability to maintain the Technical Specification required temperature of 1700 degrees F. was verified. The bench test also demonstrated that the ignitors could maintain adequate temperatures down to 10 volts. The decreased voltage applied will lessen the likelihood of ignitor failures caused by operation above design limits.

Evaluation: There are no unreviewed safety questions associated with this modification. The modification will result in more reliable performance. The modification will have no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required.

72 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-70113, Revise UFSAR Table 9-3, Nuclear Service Water flow diagrams, System Test Acceptance Criteria and System Design Basis Documents

Description: UFSAR Table 9-3, Nuclear Service Water System and Containment Ventilation System Test Acceptance Criteria Sheets, Nuclear Service Water System and Containment Ventilation System Design Basis Documents and Nuclear Service Water System Flow diagrams are being revised to reflect actual system flow conditions which were measured after modifications CN-11248 and CN-20639 were performed. After these modifications were implemented, the Containment Chilled Water System flow rates to the Upper Containment Ventilation Units (UCVUs) were balanced to prevent overcooling of the upper containment if a loss of Instrument Air System supply air occurred and the control valves failed open. Flow rates to the UCVUs were balanced per procedure PT/1(2)/A/4400/12. The new cooling water flow rates were not incorporated onto flow diagrams and other system documents at the time. The proposed changes to be made by this minor modification will incorporate nominal flow rates set under these modifications. The Containment Ventilation System Design Basis Document will be revised to show the nominal field flow rates. The Nuclear Service Water System Design Basis Document is being revised to remove statements which indicate that the Containment Coolers maintain containment temperatures within limits during design basis events since this is not true in any of the safety analyses or calculations associated with these events. UFSAR Table 9-3 and the Nuclear Service Water System Design Basis Document will be revised to approximate the Nuclear Service Water System flow rates to the UCVUs based upon the Containment Chilled Water System flow adjustments made after modifications CN-11248 and CN-20639 were implemented. UFSAR Table 9-3 and Nuclear Service Water System Design Basis Document Table 3 are also being revised to show the Lower Containment Ventilation Unit nominal flows (800 gpm) measured.

The following changes were performed under previous modifications and the 10CFR50.59 evalauations for the modifications analyze the questions of Unreviewed Safety Question applicability. The changes under this mod for these issues are editorial. The Auxiliary Building Radwaste Area Supply Unit flows will be deleted from Nuclear Service Water System Design Basis Document Table 3 since the Nuclear Service Water System was disconnected from the Unit under modification CN-10809. Test Acceptance Criteria Sheet CNTC-1574-RN-SOOI-02 will be revised to delete the Auxiliary Building Radwaste Supply Area Vent Unit and Fuel Handling Area Supply Units' flow from the Nuclear Service Water System since these units were modified under modifications CE--61290, CE-61137 and CN-10809 to remove the Nuclear Service Water System connection to these units.

Evaluation: The Containment Ventilation System is described in UFSAR Section 9.4.6. The Nuclear Service Water System is described in UFSAR Section 9.2.1. The section of the UFSAR which will be revised by this modification is UFSAR Table 9-3 which contains flow data for the various branches of the Nuclear Service Water System. The Containment Ventilation System does not provide any safety related function and is not required to mitigate the consequences of any postulated accidents. Its function is for containment temperature control for normal plant operation and normal shutdown only. The LOOP Containment Ventilation System design basis function will not be adversely affected by this modification. Technical Specification 3.6.5 describes the limits for containment

temperatures. This Technical Specifction is not affected by this modification. The Nuclear Service Water System is used during Loss of Offsite Power to provide cooling water to the containment cooling systems since the Containment Chilled Water System will not be available. The UCVU flow rates revised in UFSAR Table 9-3 and the Nuclear Service Water System Design Basis Document are estimated values based upon the current plant configuration. No credit is taken for these flow rates in any safety analyses to mitigate any design basis event. The changes made to the UCVU flow rates will not affect cooling of the reactor coolant pumps upon a loss of the Containment Chilled Water System. This evaluation will addresses the unreviewed safety question determination for modification CE-70113.

The proposed changes to the UFSAR, Design Basis Documents, Test Acceptance Criteria Sheets and Flow Diagrams for the Containment Ventilation System, the Containment Chilled Water System and Nuclear Service Water systems serve to clarify the description of present operating conditions already in place. The existing configuration of the plant was modified per modifications CN-11248 and CN-20639 and the changes to the plant were evaluated under the associated 19CFR50.59 evaluations for those modifications. The documents which will be modified per this modification (CE-70113), should have been changed under the modifications CN-11248 and CN-20639 but were apparently overlooked.

The Containment Ventilation System and the Containment Chilled Water System function together to maintain acceptable temperature limits within the confines of the Reactor Building upper and lower compartments to ensure proper operation of equipment and controls during normal plant operation and normal shutdown and for personnel access during inspection, testing, and maintenance. The Containment Ventilation System does not provide any nuclear safety related function and is not required to mitigate the consequences of any postulated accidents. Its function of containment temperature control is for normal plant operation and normal shutdown only. Nuclear Service Water System water supply to the Containment Ventilation system occurs during loss of offsite power since Containment Chilled Water cooling water is not available.

The Containment Ventilation/ Containment Chilled Water System maintains temperatures within the established containment environmental qualification limits and initial average temperatures assumed in the Design Basis Accident, LOCA and SLB analyses. This modification is only correcting plant documents to reflect the current Containment Chilled Water System configuration. This modification will not adversely affect the design basis function of the Nuclear Service Water /Containment Chilled Water Systems or the temperature limits associated with Technical Specification 3.6.5.

These changes will not affect any of the accident initiators in the SAR. The Containment Ventilation System does not provide any safety related function and as such is not required to mitigate any postulated accidents. Its function of containment temperature control is for normal plant operation and normal shutdown only. The changes to documents occurring in this modification reflect actual field conditions. The operation of these systems will not be impacted . This modification will not increase the probability of an accident previously evaluated in the UFSAR.

The Containment Ventilation System does not provide any nuclear safety related function

and is not required to mitigate the consequences of any postulated accidents. The ability of the Containment Ventilation / Containment Chilled Water System to perform its design basis function will not be affected. The number of operational ventilation units is controlled procedurally to ensure the required temperature limits are maintained. The changes made for the UCVU flow rates will not affect cooling of the reactor cooling pumps upon a loss of the Containment Chilled Water System. Therefore this modification will not increase the probability of an occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR.

There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. UFSAR Table 9-3 will be revised.

7 Type: Minor Modification **Unit:** 0

Title: Minor Modification CE-70144 Remove straightening section from Control Room Area Ventilation System air flow monitor.

Description: This modification will allow the straightening section from the Control Room Area Ventilation System air flow monitor 1CR-AFMD-4 (1VCFE5200) or 2CR-AFMD-4 (2VCFE5200) located in the supply ductwork on elevation 577' to be removed.

The design basis function of the Control Room Area Ventilation System is to ensure that the control room will remain habitable for operations personnel during and following all credible accident conditions. The system also ensures that the ambient air temperature does not exceed the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system. This function is accomplished by pressurizing the Control Room to greater than 1/8 inch water gauge with respect to all surrounding areas, filtering the outside air used for pressurization, filtering a portion of the return air from the Control Room to clean-up the Control Room environment, and by maintaining the Control Room temperature at </= 90 degrees F.

This modification will allow the straightening section from air flow monitor 1CR-AFMD-4 (1VCFE5200) or 2CR-AFMD-4 (2VCFE5200) to be removed. The straightening section is located within the airflow monitor just upstream of the airflow monitor pitot tubes and the air sample tubes. The air flow monitor is manufactured by the Air Flow Monitor Company. The design air flow through the airflow monitor is 4,000 cfm and the inlet duct is round 24 inch diameter stainless steel. The maximum velocity of air through the inlet ducting is approximately 1,325 fpm. Using the manufacturers technical data, removal of the air flow monitor straightener will reduce the air flow resistance approximately .027 in wc. The Unit 1and 2 CRA-PFT-1 fan shows that the actual resultant increase in air flow will be minimal.

This air flow monitor is used to verify one Control Room Area Ventilation System train can maintain a positive pressure of >/= 0.125 inches water gauge, relative to the adjacent areas during the pressurization mode of operation at a makeup flow rate of </= 4000 cfm per Technical Specification Surveillance Requirement 3.7.10.4. The air flow monitor will continue to provide air flow rate measurements as designed after the straightening section is removed. There is approximately five duct diameters of straight duct upstream of the air flow monitor. This is slightly less than the 7.5 duct diameters recommended by industry standards. The duct layout is such that a long radius elbow is located upstream. Since this is not a sharp transition the effect on air flow measurement will be minimal. Actual field test data was reviewed back to February 1999, this data showed that the maximum inlet air flow entering A or B outside air intakes was 2,446 cfm. This maximum air flow is much less than the allowable maximum (4,000 cfm) identified as acceptance criteria in the Technical Specification.

Evaluation: Performance testing will be completed after removal of airflow monitor straightening vanes. This testing will verify that Control Room outside inlet airflow continues to meet the acceptance criteria of Technical Specifications. There are no unreviewed safety questions associated with this minor modification. No Technical Specification changes are required. No UFSAR changes are required

75 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-70221 - Changes to UFSAR Section 6.2.2.2

Description: UFSAR Section 6.2.2.2 will be revised to correctly describe the containment emergency sump. The revision is necessary because the existing text addressing the screen assembly covers is inaccurate. Currently it is stated that the screen assembly covers are sloped and contain holes that are smaller than the fine mesh in the screen assembly. This statement is not correct. The screen assembly covers are flat and the holes are one eight of an inch in diameter (slightly larger than the 0.120 inch fine mesh opening).

Evaluation: The containment emergency sump functions as part of the Emergency Core Cooling System (ECCS) to provide a suction source for the residual heat removal pumps and containment spray pumps during the recirculation phase of an accident. There is one common ECCS suction line per train located in the pipe chase (or tunnel) area of lower containment. Each of these lines is housed within a structure that precludes foreign material larger than one-eighth inch from entering the suction lines. Fine mesh screen assemblies (mesh opening is 0. 120 inches) are installed on all sides. In addition, solid plates are installed on the tops of the screen assemblies for vortex suppression. Holes are drilled in the plate material to ensure that any air trapped by the rising water is vented and not available to be drawn into the suction piping. There is no slope on these plates, but (due to the drilled holes) sloping is not necessary to remove most if not all of the trapped air.

The cover plate vent holes are slightly larger than the opening in the fine mesh screen (0. 125" vs. 0. 120"). This difference is acceptable because one of the design criteria for the sump is to preclude particles greater than 0.125" from entering. Acceptability has been further confirmed by an Operability Evaluation. ECCS analysis assumptions relative to the total area of fine mesh screen and/or amount of this area assumed to be blocked by debris are unaffected by the presence of the vent holes. This change will not add any new vent holes or modify the existing ones. The change clarifies UFSAR text relative to the As-Built plant condition. The As-Built configuration agrees with approved design drawings. There are no unreviewed safety questions associated with this change. No Technical Specification changes are required. UFSAR Section 6.2.2.2 will be revised.

91 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-70244, Replace Valve 1ZP-33

Description: Modification CE-70244 will replace valves 1ZP-33 (Item Numbers CMV-662) with new Item Numbers DSV-031. Presently this valve is a one inch T-Type Globe valve. The replacement valve is a one inch Ball valve that is suitable for this application.

The Vacuum Priming System is used to continuously remove air from cooling water lines, condenser waterboxes and other heat exchangers that operate under a vacuum on the cooling water side of the plant. Valve 1ZP-33 is the 1A1 Main Condenser Inlet Waterbox Priming Valve outlet.

Evaluation: There is no Unreviewed Safety Question associated with modification CE-70244. The change has no effect on the probability or consequences of accidents analyzed in the UFSAR. Flow diagram CN-1598-03.01 (not a UFSAR Figure) will be revised to show the new valve design. No Technical Specification changes or UFSAR changes are required.

207 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-70245, Replace Valve 1ZP25

Description: Modification CE-70245 will replace valve 1ZP25 (Item Number 06J-601) with a new item number (DSV-031). Currently this valve is a one inch T-Type globe valve. The replacement valve is a one inch ball valve that has been evaluated to be a suitable replacement. Valve 1ZP25 is the "1A2 Main Condenser Outlet Waterbox Priming Valve Outlet".

Modification CE-70244 will replace valves 1ZP-33 (Item Numbers CMV-662) with new Item Numbers DSV-031. Presently this valve is a one inch T-Type Globe valve. The replacement valve is a one inch Ball valve that is suitable for this application.

The Vacuum Priming System is used to continuously remove air from cooling water lines, condenser waterboxes and other heat exchangers that operate under a vacuum on the cooling water side of the plant.

Evaluation: There is no Unreviewed Safety Question associated with modification CE-70245. The change has no effect on the probability or consequences of accidents analyzed in the UFSAR. Flow diagram CN-1598-03.01 (not a UFSAR Figure) will be revised to show the new valve design. No Technical Specification changes or UFSAR changes are required.

237 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-70265, "Noise Supression on Containment Pressure Control System Control Loop 1NSLP5260 and 1NSLP5180

Description: Minor Modification CNCE 70265 will add noise suppression devices to three components in the Nuclear Spray System Pump's Containment Pressure Control System (CPCS) permissive control loops. The first component (the BG relay coil) gives the permissive to start and stop the Containment Spray Pumps. This modification will also add a noise suppression device to the BF relay coil, which energizes when the Catawba CPCS spray valve permissive (1NSLP5270/1NSLP5190) is put into test. The third noise suppression device will be added to the BE current alarm, which changes states to energize the BG relay. The time delay introduced by these noise suppression devices has been evaluated for two effects. 1) Starting of the Containment Spray Pump during a containment pressure increase transient. This has been shown to have no effect on the actual start time of Containment Spray Pump due to the sequencer's longer time delay. 2) Termination of Containment Spray Pump operation during decay of containment pressure. This has been evaluated to show the noise suppression devices will make no measurable difference between the containment pressure at the time the Containment Spray Pump terminates.

Evaluation: There are no unreviewed safety questions associated with this minor modification. The activity will have no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required.

118 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-70314

Description: Spent Fuel Pool Cooling System demineralizer problems and frequent purification loop post-filter change out were experienced during the Unit 2 EOC10 refueling outage. One solution to these problems is to return the Spent Fuel Pool Cooling System purification loop pre-filters to service. Placing a pre-filter in service removes larger suspended particles in the flow stream prior to these particles entering the demineralizer. In this manner, the demineralizer is not unnecessarily contaminated and the expensive, fine filtration post-filters are clogged to a lesser extent by material carry-over.

Plant documentation will be revised to show the Spent Fuel Pool Cooling System purification loop pre-filters as normally in service. In addition, UFSAR Section 9.1.3.1.2 will be revised to state that filter change-out criteria is approximately 45 psid instead of 35 psid.

Evaluation: UFSAR Section 9.1.3 describes the design basis of the Spent Fuel Pool Cooling and Purification System, This section states:

"The Spent Fuel Pool Cooling System is designed remove heat from the spent fuel pool and maintain the purity and optical clarity and fuel handling operations. The purification loop provides an alternate means of removing impurities from either the refueling cavity/transfer canal water during refueling or the refueling water storage tank following refueling,."

To accomplish this, the Spent Fuel Pool Cooling System purification loop contains two parallel pre-filters, a demineralizer, and two parallel post-filters. Current normal system operation consists of the demineralizer in service with one post-filter and a purification loop flow rate of approximately 265 gpm. After implementation of this modification and UFSAR change, one pre-filter may be placed in service depending on plant conditions. It is anticipated that the pre-filter - demineralizer - post-filter flow path will be used mostly during refueling canal or Refueling Water Storage Tank cleanup. However, the proposed changes allow pre-filter usage during any plant condition where the purification loop is placed in service. There will be no problem achieving the required purification loop flow rate with an additional filter in the flow path. Purification loop outlet throttle valve 1(2)KF-36 is almost completely closed at 265 gpm. Opening this valve slightly will more than compensate for the extra pressure drop associated with the pre-filter.

Using a pre-filter will reduce contamination of the demineralizer and frequent post-filter change out. The proposed larger micron size filters have been previously approved for use and the appropriate notations are present on plant drawings. Parts of the Spent Fuel Pool Cooling System are Safe Shutdown related, but the purification loop components are not credited in any Safe Shutdown analysis. Therefore, the normal position changes for valves 1(2)KF-30, 31, and 38 will not adversely affect any credited Safe Shutdown functions. This modification and UFSAR change will not affect the fine filtration performed by the purification loop postfilters. Likewise, the larger DP change-out criteria on both the pre and post-filters will not affect filter performance or system integrity as these filters are rated up to 75 psid per the manufacturer. Pool cooling and pool/refueling canal optical clarity are unaffected by this change. Finally, the ability to place the FWST

in cleanup mode is unaffected.

The Spent Fuel Pool Cooling System purification loop is not considered an accident initiator in any UFSAR accident analysis.

Placing a purification loop pre-filter in service will not affect any other equipment required to cool the Spent Fuel Pool. After placing any purification loop filter in service, the filter housing is checked for leakage at system pressure to ensure that no possibility exists for slowly pumping down the Spent Fuel Pool to an unacceptable level. In addition, this condition would be detected by control room level alarm prior to going below the Technical Specification required level.

This change does not result in an increase in offsite dose for any accident previously evaluated. There is no effect on Spent Fuel Pool Level or total Spent Fuel Pool Cooling System Flow Rate.

This change will not allow performance of unanalyzed activities that may cause an accident different than what is currently evaluated. The Spent Fuel Pool Cooling System purification loop was designed with the pre-filters present, so placing one in service does not invalidate the original concept for the system, nor will it introduce any new failure modes that may lead to a currently unanalyzed accident.

There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. UFSAR Sections 9.1.3.1.2, 9.1.3.2.2.1 and Table 9- will be revised.

121 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-70349, Procedure IP/0/B/3314/038 F Revision 12 and Calculation CNC-1227.00-00-0060, Revision 2

Description: Following a modification to replace the flow instrumentation on the Containment Particulate, Iodine, and Gas Radiation Monitor with an improved accuracy device, the flow rate through the containment airborne monitor could not be established at 5 Standard Cubic Feet Per Minute (SCFM) as stated in the UFSAR. A modification was performed to revise a calculation to establish the Operator Aid Computer Delta Count Rate setpoints for the radiation monitors based upon a 3 SCFM flow rate. These setpoints are used in support of Technical Specification 3.4.15, Reactor Coolant Leakage Detection. This Technical Specification requires that a one gallon per minute reactor coolant system leak, be detected within one hour. The revision to the setpoints supports the margin of safety established in the Technical Specification bases.

Evaluation: There are no unreviewed safety questions associated with this activity. The Containment Particulate, Iodine, and Gas Radiation Monitor is not an accident initiator. This change will have no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. UFSAR Section 11.5.1.2.2.2 and UFSAR Table 5-10 will be revised.

238 **Type:** Minor Modification **Unit:** 2

Title: Minor Modification CE-70430, "Noise Supression on Containment Pressure Control System Control Loop 2NSLP5260 and 2NSLP5180

Description: Minor Modification CNCE 70430 will add noise suppression devices to three components in the Nuclear Spray System Pump's Containment Pressure Control System (CPCS) permissive control loops. The first component (the BG relay coil) gives the permissive to start and stop the Containment Spray Pumps. This modification will also add a noise suppression device to the BF relay coil, which energizes when the Catawba CPCS spray valve permissive (2NSLP5270/2NSLP5190) is put into test. The third noise suppression device will be added to the BE current alarm, which changes states to energize the BG relay. The time delay introduced by these noise suppression devices has been evaluated for two effects. 1) Starting of the Containment Spray Pump during a containment pressure increase transient. This has been shown to have no effect on the actual start time of Containment Spray Pump due to the sequencer's longer time delay. 2) Termination of Containment Spray Pump operation during decay of containment pressure. This has been evaluated to show the noise suppression devices will make no measurable difference between the containment pressure at the time the Containment Spray Pump terminates.

Evaluation: There are no unreviewed safety questions associated with this minor modification. The activity will have no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required.

149 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-70432, Revision of Drawing CNM-1211.00-1173 for Fuel Pool Ventilation System and Service Building and Warehouse Ventilation System Fire Dampers

Description: Vendor part numbers have changed for fire dampers identified on vendor darwing CNM-1211.00-1173. Minor Modification CE-70432 provides new part numbers as given by the manufacturer of the fire dampers. This vendor drawing currently shows some dampers that are nuclear safety related and some that are not nuclear safety related. The drawing does not distinguish betwen the two. This modification will identifiy the "Safety Realed Status" of each damper on the drawing.

Evaluation: There are no unreviewed safety questions associated with this minor modification. The modification provides more accurate drawing information but does not make any actual changes to the plant. No Technical Specification changes are necessary. No UFSAR changes are required.

252 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-70467, Revise flow diagram to show valve 1RY55 as non - functional

Description: Minor Modification CE-70467 will revise the Exterior Fire Protection System flow diagram to show valve 1RY55 as non -functional.

Evaluation: Valve 1RY-55 is an exterior fire loop isolation roadway box. The valve has become difficult to operate . The valve was intended to provide isolation for two old buildings that have been demolished. The supply lines to these buildings have been capped. Since the valve is located in a difficult position and serves no purpose, it will be left permanently open. This will have no effect on the Exterior Fire Protection System. The Exterior Fire Protection System is not nuclear safety related. This valve is not related to any accident analyzed in the UFSAR. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. UFSAR Figure 9-136 will be revised.

236 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-70492 and CE-70493, Fuel Pool Cooling System - Relocate differential pressure switch ports from the Auxiliary Building to the Fuel Pool Building

Description: Minor modification CE-70492 and CE-70493 will remove the honeycomb section from Fuel Building Ventilation System filtered exhaust airflow monitors 1(2)VFFE5310 and 1(2)VFFE5340. The air straightening section is made of a structure shaped like a honeycomb with small openings that are getting blocked with dirt and debris. After removal of the honeycomb section, the airflow monitor will continue to provide airflow indications. The seismic integrity of the airflow monitor will not be compromised. The Fuel Building Ventilation System airflow rate will be measured after the modification to ensure it remains within the Technical Specification limits. This modification will also relocate the low pressure port for differential pressure switches 1(2) VFPS5360 and 1(2) VFPS5480. This port will be moved from the Auxiliary Building and placed in the Fuel Pool Area. Relocation of the low pressure port in the Fuel Pool Area will provide a reliable pressure signal for operation of the fuel pool supply fan.

Evaluation: There are no unreviewed safety questions associated with this minor modification. The activity will have no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required.

158 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-70514, Removal of Air Flow Monitor (1VPFE5200) Straightener Section

Description: Modification CE-70514 will remove the honeycomb section from a Containment Purge System exhaust airflow monitor (lVPFE5200 or 1CPE AFMD-1). The air straightening section is made of a structure shaped like a honeycomb with small openings that are getting blocked with dirt and debris. After removal of the honeycomb section, the airflow monitor will continue to provide airflow indications. The seismic integrity of the airflow monitor will not be compromised. The Containment Purge System airflow rate will be measured after the modification to ensure it remains within the Technical Specification limits.

Evaluation: There are no unreviewed safety questions associated with this modification This airflow monitor is not nuclear safety related. It is not an accident initiator. The monitor will continue to balance the airflow rates during fuel handling operations. Structural integrity of the ductwork that is used for removal of air from the containment during postulated fuel handling accident will not be affected. This modification will not affect the probability or consequences of accidents described in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required.

241 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-70522, "Hotwell Pump 1B and 1C Cartridge-Type Mechanical Seal"

Description: Minor Modification CE-70522 involves replacement of the Hotwell Pump 1B and 1C mechanical seals from component type seals to cartridge type seals. The replacement seal introduces no new failure modes and are expected to have a reliability equal to or greater than the original seals. Installation of the new seals requires some minor seal flush piping changes that require flow diagram revisions.

Evaluation: There are no unreviewed safety questions associated with this minor modification. This modification will have no effect on the probability or consequences of accidents evaluated in the UFSAR. No Technical Specification changes are required. UFSAR Figure 10-18 will be revised.

166 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-70535, Replace Valves 1LD-11 and 1LD-12 with new 02D-2002 Valves

Description: Valves 1LD-11 and 1LD-12 are currently carbon steel plug valves. These valves are a part of the Diesel Generator Lube Oil System. These valves have developed seal leaks, but no spare parts or identical spare valves are available. Therefore, the valves will be replaced with new carbon steel ball valves, item number 02D-2002. All affected station documents will be revised to reflect this new information.

Evaluation: There are no unreviewed safety questions associated with this valve replacement. The new valves are functionally equivalent to the ones they are replacing. Any differences between the old valves and the replacement valves were evaluated and approved by Engineering. No Technical Specification changes are required. UFSAR Figure 9-185 (a piping flow diagram) will be revised.

243 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-70540, "Guidelines for tubing to HVAC instruments and equipment"

Description: This modification will provide guidelines for repair and replacement of copper/stainless tubing to Heating, Ventilation, and Air Conditioning (HVAC) instruments and equipment.

Evaluation: These HVAC tubing guidelines are being provided for tubing routed to HVAC equipment and instruments. HVAC tubing was initially installed as follows:

Nuclear safety related differential pressure switches across nuclear safety related fans by the HVAC Controls Subcontractor (MCC Powers). MCC Powers had their own QA program and installed the tubing as nuclear safety related per that program.

Nuclear safety related differential pressure transmitters for total carbon pressure drop, installed by the filter manufacturer (MSA). This manufacturer installed the tubing as part of a nuclear safety related filter unit, per their QA program.

Differential Pressure gauges across nuclear safety related carbon filter unit components such as HEPA filters, Carbon Bed and Moisture Separator sections. These pressure gauges were installed by the filter manufacturer (MSA). This manufacturer installed the tubing to these gauges as part of a nuclear safety related filter unit, per their QA program.

Copper tubing for essential components of the Control Room Chilled Water System chiller. This tubing was installed by Carrier as a part of their QA program.

Installed copper tubing to the above applications is considered nuclear safety related under the manufacturer's QA program.

The Guidelines provided by this modification will ensure that HVAC tubing or associated instruments that attach to this tubing continue to operate as designed to support the design basis for the HVAC system affected. Seismic concerns are addressed in the modification by requiring the tubing to be reinstalled as originally installed or, if changes are necessary, by requiring a modification to address new routing, new material (stainless) and/or hanger loads. Failure of the tubing is addressed by requiring similar tubing to be used for any changes.

The following systems have tubing routed to HVAC safety related equipment or instruments:

Auxiliary Building Ventilation System
Control Room Ventilation System
Diesel Room Ventilation System
Annulus Ventilation System
Fuel Building Ventilation System
Nuclear Service Water Pump Structure Ventilation System
Control Room Chilled Water System

The Auxiliary Building Ventilation System provides ventilation for the Auxiliary

Building during normal operation and during a design basis accident. This system maintains the ECCS pump rooms at a negative pressure with respect to the adjacent areas. Potentially contaminated air exhaust from the ECCS pump rooms is filtered prior to release to the environment.

The Control Room Ventilation System provides ventilation for the Control Room. This system maintains the Control Room at a positive pressure relative to adjacent areas. Potentially contaminated air pulled in from the environment is filtered by the system prior to allowing air to enter the Control Room.

The Diesel Building Ventilation System is designed to provide ventilation for the Diesel Building both during normal operation and when the diesel is operating.

The Annulus Ventilation System is designed to maintain the Annulus at a negative pressure with respect to its adjacent areas. Potentially contaminated air exhausted from the Annulus is filtered prior to release to the environment.

The Fuel Building Ventilation System is designed to provide ventilation during normal operation and to maintain a negative pressure in the Fuel Building during the movement of fuel within the pool. Potentially contaminated air exhausted from the Fuel Building is filtered prior to release to the environment.

The Nuclear Service Water Pump Structure Ventilation System is designed to provide ventilation for the facility both during normal operation and during a DBA.

The Control Room Chilled Water System is designed to provide chilled water to the Control Room Ventilation System .

The tubing to HVAC equipment or instruments is not an accident initiator as described in the UFSAR. These guidelines will allow HVAC tubing to continue to function properly to allow the filter units to operate in either the filter or the bypass alignment. These filters fail to the filtered alignment during a design basis accident and will continue to filter as required. The tubing for HVAC instruments will provide fan status lights or filter differential pressure lights. Failure of this tubing will not affect the operation of the ventilation system. Therefore, this modification will not increase the probability of occurrence of an accident previously evaluated in the UFSAR. Implementation of this modification will not cause the plant's nuclear safety related ventilation systems to be operated outside of their design parameters. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. No UFSAR changes are required.

182 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-70563, Replace Valves 1NV-323, 1NV-333, and 1NV-335 and delete valves 1NB-889, 1NB-895, and 1NV-896

Description: Minor Modification CE-70563 will replace valves 1NV-323, 1NV-333, and 1NV-335 with a new valve of a different item number and delete the packing isolation leakoff valves 1NB-889, 1NB-895, and 1NV-896. The leafoff lines and isolation valves are no longer required since the new valves are bellows-sealed.

Evaluation: There is no unreviewed safety question associated with this modification. The replacemet of the valves and the deletion of the leafoff lines and isolation valves will have no significant effect on the performance of the system. This modification will have no effect on the probability or consequences of accidents evaluated in the UFSAR. No Technical Specification changes are required. UFSAR Figures 9-91 and 9-200 will be revised.

194 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-70610, Replace piping upstream of valve 1RN865 and add valve 1RN-F61

Description: Minor Modification CE-70610 will replace a portion of piping between valve 1RN865 and the Nuclear Service Water Pump Structure pit wall. The portion of the piping below the waterline in the "B" pit has degraded, is leaking, and must be replaced. The existing carbon steel piping below the water line will be replaced with stainless steel piping. Valve 1RNF61 will be permanently added as a means to isolate the system during this piping replacement.

Evaluation: The changes per modification CE-70610 are considered a like for like replacement. After the replacement of the piping, Valve 1RN-F61 will be permanently gagged open and will become a passive portion of the Nuclear Service Water System.

There are no unreviewed safety questions associated with this modification. No changes to the Technical Specifications are required. UFSAR Figure 9-24 will be revised.

226 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-70625, Replace valve 1RC55

Description: Minor Modification CE-70625 will replace valve 1RC-55 (Item Number CMV-028) with new Item Number CMV-293. Presently this valve is a 3/4" gate valve. The replacement valve is a 3/4" ball valve that is suitable for this application. All affected drawings will be revised to reflect this new information.

The Condenser Circulating Water System supplies cooling water to the main and feedwater pump turbine condensers to condense the turbine exhaust steam. 1RC-55 is an isolation valve for the 1B Condenser Circulating Water System Pump Bearing Cooler and Seal water flow control inlet.

Evaluation: Valve 1RC-55 will continue to provide all requirements necessary for plant operation and safety once it is replaced. There are no Unreviewed Safety Questions associated with Minor Modiification CE-70625. Flow diagram CN 1604-1.0 (not an FSAR Figure) will be revised to show the new valve design. No Technical Specification changes are required. No UFSAR changes are required.

242 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-70687, "Revise Documentation for the Corrosion Test Stand"

Description: The piping within valves 1RL459, 1RL460, and 1RL461 has been devoted to the Raw Water Corrosion Test Stand (corrosion Test Header). The arrangement of the piping and components between these valves has periodically changed due to the need to change testing configurations. This modification will revise the existing documentation of piping between 1RL459, 1RL460, and 1RL461 to better facilitate changes to the test stand. Configuration and operation of the test stand within valves 1RL459, 1RL460, and 1RL461 is controlled by procedure.

Evaluation: There are no unreviewed safety questions associated with this minor modification. This modification will have no effect on the probability or consequences of accidents evaluated in the UFSAR since the affected piping does not perform any safety function. No Technical Specification changes are required. UFSAR Figure 9-66 will be revised.

164 **Type:** Miscellaneous Items **Unit:** 0

Title: Administrative Controls Associated with the Degradation of the ECCS Pump Room Pressure Boundary, Revision 0

Description: The "Administrative Controls Associated with the Degradation of the ECCS Pump Room Pressure Boundary" contains the administrative controls necessary to rapidly re-establish the pressure boundary integrity of the ECCS Pump Rooms. Modifications and/or repairs to the ECCS Pump Room pressure boundary can adversely affect the capability to maintain the pump rooms at a negative pressure relative to the surrounding areas. The ability to maintain a negative pressure within the pump rooms is a design basis function of the Auxiliary Building Ventilation System.

Evaluation: The Auxiliary Building Ventilation System functions to provide normal ventilation requirements within the Auxiliary Building and, during a Design Basis Accident, the Auxiliary Building Ventilation System functions to maintain a negative pressure within the ECCS Pump Rooms. This function is addressed in Technical Specification Surveillance Requirement 3.7.12.4. The ECCS Pump Room pressure boundary (walls, floor, roof, doors, some Auxiliary Building Ventilation System ductwork, and fire/pressure seals) is essential to maintaining this negative pressure. Certain plant activities may require the ECCS Pump Room pressure boundary to be breached. In order to address temporary degradation of the pressure boundary, Technical Specification Amendments No. 187 (Unit 1) and No. 180 (Unit 2) were obtained. Under these amendments the ECCS Pump Room pressure boundary can be degraded for up to 24 hours provided that administrative controls are in place to rapidly re-establish pressure boundary integrity if needed. The administrative controls are contained in the "Administrative Controls Associated with the Degradation of the ECCS Pump Room Pressure Boundary". Since the Technical Specifications allow for the boundary to be degraded, no unreviewed safety questions exist. However, there are several issues that require further discussion and clarification on the pressure boundary. These issues are discussed below.

The SER issued for Technical Specification Amendments No. 187 (Unit 1) and No. 180 (Unit 2) states that during the period when the ECCS Pump Room pressure boundary is inoperable, "appropriate compensatory measures, consistent with the intent" of 10CFR100 will be utilized to protect plant personnel. Neither the "appropriate compensatory measures" nor how the "intent" of meeting 10CFR100 are defined. These issues are addressed in the "Administrative Controls Associated with the Degradation of the ECCS Pump Room Pressure Boundary" by requiring any ECCS Pump Room pressure boundary opening to have a designated individual, who is in constant communication with the control room, stationed near the opening with a method of rapidly sealing the opening upon notification. Rapidly is defined as within ten minutes of several plant events including a Safety Injection signal. Ten minutes is an acceptable time limit since this ensures that, assuming each ECCS pump is at maximum flow and the Refueling Water Storage Tank (RWST) is at its Technical Specification minimum limit, the breach will be closed prior to the initiation of containment sump recirculation. Containment sump recirculation is the earliest time a pipe rupture or high radiation in the pump rooms becomes a concern. Adequate controls have been established in the administrative controls to ensure that the door is closed within ten minutes of a safety injection signal or other significant plant event. Thus, the "intent" of meeting 10CFR100 is clearly satisfied.

It should also be noted that, except for the effect on the ECCS Pump Rooms, the Auxiliary Building Ventilation System will be unaffected by a breach in the pressure boundary and will continue to provide ventilation for the Auxiliary Building.

For work that involves drilling additional penetrations in the ECCS Pump Room pressure boundary, Modification Engineering ensures that the seismic qualification of the pressure boundary is not affected. Maintenance work associated with the ECCS Pump Room pressure boundary usually involves work on the doors or opening a door to pass cable or some other items through the door. All the doors to the ECCS Pump Rooms are seismically qualified in the open and the closed position. Therefore, if the plant were to sustain a SSE event while the door was open it would still be capable of being closed, thus, restoring the ECCS Pump Room pressure boundary. It is not assumed that debris from adjacent areas would fall in the ECCS Pump Room doorways during a SSE and prevent the doors from closing. This assumption is credible since all the ECCS Pump Rooms are within the Auxiliary Building, which is a Seismic Category 1 structure; and ductwork, piping and cable trays located near the pump rooms are seismically supported.

Another issue is whether or not the breaching of an ECCS Pump Room pressure boundary is subject to single active failure criteria. A review of the present system design shows that these administrative controls/compensatory actions will not affect the capability of the Auxiliary Building Ventilation System to function assuming a single active failure. This is based on the recognition that certain aspects of the ECCS Pump Room pressure boundary are inherently not single failure proof. For example, Catawba is not designed with vestibules at any door leading to an ECCS Pump Room, nor is it designed with redundant walls around the pressure envelope. This lack of multiple failure barriers is acceptable based on the nature of the barriers; i.e., they are either passive components such as walls, or they are simple mechanical devices used as they were designed, such as doors. It is also recognized that opening the pressure boundary (such as having a door held open and having to close it) constitutes changing the ECCS Pump Room pressure barrier from a passive function (a closed door) to an active function (having to close the door). This is acknowledged by the Technical Specifications that allow the pressure boundary to be degraded for up to 24 hours if a method of rapidly re-establishing the pressure boundary exists. Also, the administrative controls ensure that the work crew has the means to restore the integrity of the pressure boundary upon notification by the Control Room. It is assumed that the work crew can perform this task without making an error based on the clear, simple instructions given in the administrative controls and the pre-job brief that will be held prior to using them. This ensures all actions are clearly understood and adequate processes are in place to fulfill the actions. Thus, the loss of the pressure boundary is considered to be previously reviewed by the NRC and the requirement to re-establish the pressure boundary under administrative controls is not subject to single failure criteria.

These administrative controls/compensatory actions direct the Control Room SRO, or designee, to notify the work crew to restore the ECCS Pump Room pressure boundary integrity so that the capability to pressurize the pump rooms is restored. This guidance is consistent with Technical Specification 3.7.12 Action B. The effect of using these administrative controls/compensatory actions on each of the ECCS Pump Room design basis functions was evaluated. These design basis functions include: ventilation pressure boundary, radiation boundary, pipe break boundary, and flood boundary.

There are no unreviewed safety questions associated with these administrative controls. No further changes to the Technical Specifications are required. No UFSAR changes are required.

163 **Type:** Miscellaneous Items **Unit:** 0

Title: Administrative Controls Associated with the Degradation of the Control Room Pressure Boundary, Revision 12

Description: As described in the Administrative Controls Associated with the Degradation of the Control Room Pressure Boundary, modifications and/or repairs to the Control Room pressure boundary can adversely affect the capability to pressurize the Control Room relative to the Service Building, Auxiliary Building, and outside areas adjacent to the Control Room. Pressurization of the Control Room is a design basis function that is necessary to protect the Control Room operators from the effects of certain design basis accidents.

Evaluation: The Control Room Area Ventilation System provides ventilation to the Control Room and the Control Room area. This system maintains the Control Room proper at a positive pressure per Surveillance Requirement 3.7.10.4. The Control Room pressure boundary (walls, floor, roof, doors, some system ductwork and dampers, and fire/pressure seals) is essential to maintaining this positive pressure. Technical Specification Amendments No. 187 (Unit 1) and No. 180 (Unit 2) allow the Control Room pressure boundary to be degraded for up to 24 hours provided that administrative controls are in place to rapidly re-establish pressure boundary integrity if needed. The administrative controls are contained in the Administrative Controls Associated with the Degradation of the Control Room Pressure Boundary and since the Technical Specifications allow for the boundary to be degraded, no unreviewed safety questions exist. There are, however, two issues that require further discussion on the pressure boundary - the definition of "rapidly" and how the "intent" of GDC 19 is met.

The SER issued for Technical Specification Amendments No. 187 (Unit 1) and No. 180 (Unit 2) states that during the period that the Control Room pressure boundary is inoperable, "appropriate compensatory measures, consistent with the intent" of GDC 19 will be utilized to protect operators and other plant personnel. Neither the "appropriate compensatory measures"nor how the "intent" of meeting GDC 19 and 10 CFR Part 100 are defined. These issues are addressed in the Administrative Controls Associated with the Degradation of the Control Room Pressure Boundary by requiring any Control Room pressure boundary opening to have a designated individual, who is in constant communication with the Control Room, stationed near the opening with a method of rapidly sealing the opening upon notification. Rapidly is defined as within five minutes of notification. Five minutes was chosen because the Catawba dose analysis, along with the applicable assumptions listed in the "Initial Conditions" section of the administrative controls, accounts for not having the Control Room pressurized for the first five minutes of an accident and still shows that the operator doses are within GDC 19 limits. Thus, the "intent" of meeting GDC 19 is clearly satisfied. Additionally, the five minute time period represents less than 0.4% of the allowable Technical Specification time which is assumed to meet the intent of the term "rapidly." It should also be noted that five minutes represents a time prior to sump recirculation which would minimize radiological hazards to operators and other station personnel. Thus, there are no unreviewed safety questions created by degrading the Control Room pressure boundary under these administrative controls.

It should be noted that these administrative controls were previously issued under the title

Compensatory Action Instructions Modification and/or Repair of Control Room Pressure Boundary. The title was changed to align more closely with the wording ("administrative controls") used in the Technical Specifications. One other change that was made from previous revisions to the administrative controls/compensatory action was the deletion of the initial condition for an Iodine Protection Factor (IPF) greater than or equal to 80. Historical data showed that the IPF for Catawba was always well above 80 so it was not necessary to include this parameter as an initial condition any longer.

Per Section 6.4.1 of the Catawba UFSAR, the design bases of the habitability system for the Control Room includes:

1. The capability to withstand a safe shutdown earthquake (SSE),

2. The capability to function properly following any single active failure,

3. The capability to function during a design basis tornado,

4. The capability to detect and limit concentrations of chlorine gas as specified in Regulatory Guide 1.95 or products of combustion from entering the Control Room,

5. The capability to shield Control Room operators from radiation sources,

6. The capability to detect and limit the introduction of airborne radioactive contamination into the Control Room such that exposure to personnel will not exceed the limits specified in GDC 19

7. The capability to permit safe shutdown of the plant from the Control Room following a Loss of Coolant Accident.

The Control Room Ventilation System in conjunction with the Control Room pressure boundary also serves to ensure that the Control Room environment is maintained consistent with the Environmental Qualification parameters of equipment contained within the Control Room. No postulated pressure boundary work would significantly affect the heat load on the Control Room Ventilation Chiller. Technical Specification limits for Control Room temperature will be satisfied at all times these administrative controls are in place in accordance with Technical Specification 3.7.11.

These administrative controls/compensatory actions direct the Control Room SRO, or designee, to notify the work crew to restore the Control Room pressure boundary integrity so that the capability to pressurize the Control Room is restored. This guidance is consistent with Technical Specification 3.7.10 Action B.

There are no unreviewed safety questions associated with these administrative controls. No further changes to the Technical Specifications are required. No UFSAR changes are required.

232 **Type:** Miscellaneous Items **Unit:** 0

Title: Bases for Technical Specification 3.4.13, "RCS Operational Leakage" Surveillance Requirement 3.4.13.1

Description: The Bases for Technical Specification 3.4.13, "RCS Operational Leakage" Surveillance Requirement 3.4.13.1 is being revised. This SR requires the performance, every 72 hours, of a Reactor Coolant System water inventory balance in order to verify that Reactor Coolant System operational leakage is within limits. The Bases change clarifies that the volumetric calculation of unidentified leakage and identified leakage is based on a density at room temperature of 77 degrees F and that the volumetric calculation of primary to secondary leakage is based on a density at operating Reactor Coolant System temperature of 585 degrees F.

Evaluation: There are no Unreviewed Safety Questions associated with this change. No Technical Specification changes are required. No UFSAR changes are required. The manner in which the plant and its accident mitigating equipment are designed, operated, and maintained was not affected by this change.

138 **Type:** Miscellaneous Items **Unit:** 0

Title: Bases for Technical Specification 3.8.1 "AC Sources - Operating"

Description: The description of the connection between the unit switchyard and the transmission network in the Background Bases for Tech Spec 3.8. 1, "AC Sources - Operating" is being deleted. This will be applicable to Technical Specification 3.8.2, "AC Sources - Shutdown" also, because the bases for that Technical Specification references these bases. A change is required to UFSAR Section 8.2.1.2, "Utility Grid and Switchyard Interconnections" to clarify that the switchyard is part of the primary transmission network.

Evaluation: This change affects the two offsite power sources required by Technical Specification 3.8. 1, UFSAR Section 8.2 and 10 CFR 50 Appendix A, GDC 17. There is no physical change being made to the plant, nor is there any change being made to the operation or testing of the sources. For these reasons, the change has no effect on any design basis accidents or any failure modes of these circuits. The design basis accidents researched were a Loss of External Load (UFSAR 15.2.2) and Loss of Non-Emergency AC Power to the Station Auxiliaries (UFSAR 15.2.6). The change could not cause any new or different system interactions.

There are no unreviewed safety questions associated with this UFSAR Change. No Technical Specification changes are required. UFSAR Section 8.2.1.1 will be revised .

143 **Type:** Miscellaneous Items **Unit:** 1

Title: C1C12 Reload Safety Analysis (Calculation CNC-1552.08-00-0295 - through Revision 3)

Description: A safety evaluation is performed for the Catawba Nuclear Station Unit 1, Cycle 12 (C1C12) core reload in calculation file CNC-1552.08-00-0295. The effect of any other plant changes which might be made concurrent with the refueling outage are not addressed in the calculation.

The C1C12 Reload Design Safety Analysis Review (REDSAR), performed in accordance with the Nuclear Engineering Division workplace procedure NE-102, "Workplace Procedure for Nuclear Fuel Management", serves as the safety review for the Unreviewed Safety Question Evaluation. The Nuclear Design and Reactor Support (NDRS) section of the REDSAR checklist indicates the need to further evaluate the power shapes at the limiting statepoints for the steam line break and dropped rod departure from nucleate boiling (DNB) evaluations. Also identified in the REDSAR checklist for further evaluation is the BOC prompt neutron lifetime for the set of accident analyses in which that assumption is a bounding value. An additional parameter, the Mode 5 ratio for boron dilution, will be satisfied by increasing the shutdown boron concentration in the C1C12 SOR. These evaluations, documented in CNC-1552.08-00-0295, show that the Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analyses remain bounding with respect to C1C12 safety analysis physics parameters.

The C1C12 core reload is similar to past cycle core designs, with a design generated using NRC approved methods. Additionally, no Technical Specification changes specifically related to the operation of the new C1C12 core are required. The unreviewed safety question evaluation resulted in the conclusion that there are no unreviewed safety questions concerning the C1C12 core reload.

An evaluation of an extension of C1C12 from 480 to 495 EFPD (± 10 EFPD) resulted in the identification of additional parameters that require further evaluation. The Nuclear Design and Reactor Support (NDRS) section of the REDSAR checklist indicates the need to further evaluate the power shape at the limiting statepoint for the steam line break departure from nucleate boiling (DNB) evaluation. Also identified in the REDSAR checklist for further evaluation is the EOC minimum fuel average temperature for the set of accident analyses in which that assumption is a bounding value. These evaluations, documented in CNC1552.08-00-0295, show that the Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analyses remain bounding with respect to C1C12 safety analysis physics parameters.

A safety evaluation of the Selected Licensee Commitment 16.9-12 Bases revision to the boric acid tank additional margin is also performed in calculation file CNC-1552.08-00-0295. The revision is required to ensure sufficient borated water exists in the Boric Acid Tank to borate to 1.3% shutdown in going from HFP to 200 degrees F. The unreviewed safety question evaluation resulted in the conclusion that there are no unreviewed safety questions concerning the Selected Licensee Commitment 16.9-12 Bases revision. No Technical Specification changes are required. No UFSAR changes are required.

Evaluation: There are no unreviewed safety questions associated with the Catawba Unit 1 Fuel Cycle 12 (C1C12) Reload Safety Analysis (Calculation CNC-1552.08-00-0295 - through

Revision 3). No Technical Specification changes are required. No UFSAR changes are required.

196 **Type:** Miscellaneous Items **Unit:** 2

Title: C2C11 Reload (Calculation CNC-1552.08-00-0311)

Description: A safety evaluation is performed for the Catawba Nuclear Station Unit 2, Cycle 11 (C2C11) core reload in calculation file CNC-1552.08-00-0311. The effect of any other plant changes made concurrent with the refueling outage are not addressed in the calculation.

The C2C11 Reload Design Safety Analysis Review (REDSAR), performed in accordance with Nuclear Engineering Division workplace procedure NE-102, "Workplace Procedure for Nuclear Fuel Management", serves as the safety review for the unreviewed safety question evaluation. The SAPP section of the REDSAR checklist indicates the need to further evaluate the prompt neutron lifetime, the power shapes at the limiting statepoint for the steam line break event, the maximum dropped rod worths and axial power shapes for the dropped rod event, and the total peaking factor at BOC, HZP for the rod ejection accident. These evaluations, documented in CNC-1552.08-00-031 1, confirm the Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analyses remain bounding with respect to the C2C11 safety analysis physics parameters. The safety analysis physics parameters method is described in topical report DPC-NE-3000-PA. The transition core analyses for the RFA fuel type are performed according to DPC-NE-2009-PA.

Evaluation: The C2C11 core reload is similar to past cycle core design, with a design generated using NRC approved methods. Additionally, no Technical Specification changes specifically related to the operation of the C2C11 core are required. The unreviewed safety question evaluation concludes that there are no unreviewed safety questions concerning the C2C11 core reload. No UFSAR changes are required.

62 Type: Miscellaneous Items **Unit:** 0

Title: Calculation CNC-1553.12-00-0011 (Criticality Aspects of RFA Fuel)

Description: Calculation CNC-1553.12-00-0011 (Criticality Aspects of RFA Fuel) addresses the unreviewed safety question analysis for evaluation for the use of the new Westinghouse Robust Fuel Assembly (RFA) design at Catawba. The evaluation considers the criticality aspects of the new fuel design outside the reactor.

The Westinghouse RFA fuel design is a new design to be used at Catawba in future reloads. The specific design to be used at Duke Power is based on the Westinghouse Standard (STD) fuel design. Several changes have been made to the currently analyzed STD design for the RFA design. Listed below are the key design features of the RFA fuel design.

1. ZIRLO Clad Fuel Rod
2. ZIRLO, Guide Thimbles, Instrumentation Tubes and Mid-Grids (both structural and intermediate flow mixing)
3. 0.374 inch Fuel Rod outside diameter; 0.329 inch Fuel Rod inside diameter; 0.3225 inch Fuel Pellet outside diameter
4. Fuel Pellet Theoretical Density 95%
5. Zirconium Diboride Integral Fuel Burnable Absorbers (IFBAs)
6. Mid-Enriched Annular Axial Blanket Pellets
7. High Burnup Fuel Skeleton
8. Debris Filter Bottom Nozzle
9. Increased guide Thimble and Instrumentation Tube outside diameter (0.482 inch)
10. Modified Low Pressure Drop (MLPD) Structural Mid-grids
11. Modified Intermediate Flow Mixing (MEFM) Grids
12. Pre-Oxide Coating on Bottom of Fuel Rods
13. Use of Protective Bottom End Grid (P-Grid) with Longer Fuel Rod End Plug
14. Quick Disconnect Top Nozzle
15. Fuel Rods Positioned on Bottom End Nozzle

The ZIRLO material in items 1 and 2 is identical to the current zircaloy material used except for some minor compositional changes made to enhance corrosion behavior. The major difference in the composition of the ZIRLO is the addition of niobium. However, the changes made to this material have no noticeable effect in terms of the criticality analysis. Hence the standard compositions of zircaloy used in past criticality analyses are valid for the ZIRLO material.

The fuel rod outside diameter, inside diameter and pellet outside diameter in item 3 are identical to the STD fuel design already analyzed. This fuel rod information is modeled explicitly in the criticality calculations.

The fuel density in item 4 is what is used in the current Westinghouse fuel designs. However, it should be noted a value of 95.5% TD is used as the nominal value in the criticality calculations. Uncertainty calculations account for the variance in the actual TD

about the nominal value. This is the only difference between the RFA design and the current STD design.

IFBA rods in item 5 are a poison material integral to the fuel assembly that may be taken credit of in the criticality analysis. At this time, the poison affect of the IFBA coating is conservatively ignored in the criticality analysis for RFA fuel at Catawba. Also, since this was not considered in the current licensing basis, it was decided not to take credit for it in this analysis.

The enriched annular axial blanket fuel pellets in item 6 are ignored in the criticality analysis. This is consistent with the assumption in the current analysis that ignores the solid natural U axial blankets. Instead, the criticality analysis conservatively assumes the fuel rod to be the same enrichment for the full length.

The high burnup skeleton in item 7 does not affect the criticality analysis since this is a hardware related component that is not included in the criticality model.

The debris filter bottom nozzle in item 8 does not affect the criticality analysis since this is a hardware related component that is not included in the criticality model.

The GT and IT dimensions in item 9 are identical to the STD fuel design already analyzed. This assembly information is modeled explicitly in the criticality calculations.

The design features of the grids in items 10, 11 and 13 do not affect the criticality analysis since it is standard criticality analysis practice to ignore the grids since they are not a full length structural element.

The oxide coating on the bottom of the fuel rods in item 12 does not affect the criticality analysis since the thin oxide layer does not affect the reactivity of the fuel rod.

The quick disconnect top nozzle in item 14 does not affect the criticality analysis since this is a hardware related component that is not included in the criticality model.

The fuel rod repositioning in item 15 does not affect the criticality analysis since this does not result in a significant shift in the location of the active fuel stack. Also, it should be noted that only a two dimensional model is used in the criticality analysis as it is assumed to be conservative.

After a careful review of the RFA fuel design features, most of these features are not included in the criticality model. Of the ones that are included (items 3, 4 and 9) only the TD represents a change compared to the current STD fuel design.

The impacts of these design features on the criticality analysis is implicitly included in the analysis by including the design information in the models.

Evaluation: The only accidents related to the criticality analysis outside the reactor are misloading a fuel assembly in the spent fuel pool, heavy load drop over the spent fuel pool and loss of spent fuel pool cooling. The RFA will be identified in the same manner as other fuel designs and its movement will be controlled by the same procedures as previous fuel designs. The control of heavy loads over the spent fuel pool will not change as a result of

implementing a new fuel assembly design. The loss of spent fuel pool cooling is not affected by the type of fuel stored in the pool. Therefore, the design and text changes evaluated here do not increase the probability of occurrence of an accident previously evaluated in the UFSAR.

The use of RFA fuel in areas outside the reactor will not affect any scenarios that could lead to a malfunction of equipment important to safety. The RFA design is specifically designed to reduce or eliminate the possibility of certain types of fuel failures. The fuel handling interfaces for RFA fuel are designed to be compatible with current designs and existing equipment, so the probability of a malfunction of any equipment related to fuel handling would not change. The design changes being implemented for the new RFA fuel design were developed to reduce the possibility of known failures (e.g. high burnup fuel skeleton for incomplete rod insertion). The probability of other malfunctions not addressed by any of the design changes for RFA fuel remains the same. Therefore, it is concluded that the use of RFA fuel as it relates to criticality control outside the reactor does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR.

There is no unreviewed safety question associated with the use of RFA fuel in areas outside the reactor. No technical specification changes are required. UFSAR Sections 9.1.1.3, 9.1.2.3.1.2, 4.3.2.6.1, 4.3.2.6.1 will be revised.

12 **Type:** Miscellaneous Items **Unit:** 0

Title: Calculation CNC-1553.26-00-0254, Westinghouse Robust Fuel Assembly Design

Description: This 10CFR50.59 evaluation addresses the Core Mechanical and Thermal/Hydraulic issues associated with the implementation of the Westinghouse Robust Fuel Assembly (RFA) design at Catawba Nuclear Station beginning with Unit 2 Cycle 11. The RFA design is similar to the Vantage+ Standard Westinghouse design, but contains several additional features to enhance reliability and performance. These consist of:

1. Thicker Guide Thimbles to improve straightness and minimize resistance to control rod insertion.
2. Modified intermediate spacer grids to minimize potential fuel rod grid to rod fretting and pressure drop.
3. Three additional intermediate flow mixing grids in the upper half of the assembly to improve thermal hydraulic (DNB) performance.
4. Use of an additional bottom end protective grid and elongated bottom fuel rod end grid for added debris fretting protection.
5. Use of a pre-oxide coating on the bottom 6 inches of fuel rods for added debris-fretting protection.
6. Fuel rods positioned on the bottom nozzle to minimize internal compressive loads on the guide thimbles (promote straightness).

Evaluation: This evaluation addressed all potential safety issues and licensing basis requirements relative to Core Mechanical and Thermal Hydraulic issues and concluded that there are no unreviewed safety questions associated with implementing this design. No Technical Specification changes are required. This evaluation utilizes results from a Duke RFA Functionality calculation, various Westinghouse analyses and test reports, and a recently NRC approved Topical Report addressing Duke reload methodology for RFA Fuel at the Catawba Nuclear Station. Numerous UFSAR changes are required as the result of this fuel change.

52 **Type:** Miscellaneous Items **Unit:** 0

Title: Catawba Inservice Testing Manual Revision 25

Description: The Catawba Nuclear Station Inservice Testing Program is being changed to allow removal of the open function from the testing program for valves 1(2)NI-125 and 1(2)NI-129. Valves NI-125 and NI-129 are located on the Residual Heat Removal Pump discharge header to Reactor Coolant System Loop C and B, respectively, hot leg. These valves are closed during normal operation to prevent diversion of reactor coolant to the Residual Heat Removal System if upstream check valves NI-126 and NI-134, respectively, leak. Valves NI-125 and NI-129 open to pass hot leg recirculation flow from the Residual Heat Removal System following a Safety Injection signal, if both trains of Safety Injection pumps fail and are unable to deliver flow to the hot legs. If the Safety Injection Pumps are utilized for hot leg recirculation flow, these valves remain closed to prevent diversion flow to the Residual Heat Removal System. These valves are also classified as Pressure Isolation Valves per Technical Specification 3.4.14 and UFSAR Table 5-41, "Reactor Coolant System Pressure Isolation".

The safety related function of the Residual Heat Removal System is to remove residual heat from the Reactor Coolant System. The Residual Heat Removal System transfers thermal energy from the Reactor Coolant System to the Component Cooling System during the second phase of unit cooldown. The Residual Heat Removal System transfers decay heat, stored (sensible) heat and other residual heat from the Reactor Coolant System at a rate such that design limits of the Reactor Coolant System boundary are not exceeded. The Residual Heat Removal flow path for this safety function is from the Residual Heat Removal System suction connections on hot legs B and C, through the Residual Heat Removal Pumps and Heat Exchangers, returning cooled reactor coolant to the coldlegs. Hot leg discharge lines, including check valves NI-125 and NI-129, are not included in this flow path. Once the Reactor Coolant System pressure and temperature exceed 350 degrees F and 385 psig, the Residual Heat Removal System is secured. The Residual Heat Removal System also functions as part of the Emergency Core Cooling Systems (ECCS).

The safety related function of the Safety Injection System is to provide emergency core cooling in order to prevent unacceptable fuel damage and to ensure that the core remains intact during all phases of a Design Basis Event. The Safety Injection System along with the Residual Heat Removal System, Chemical and Volume Control System and Refueling Water System is part of the ECCS. During normal plant operation the Safety Injection System is in standby readiness.

The ECCS Systems (Safety Injection System, Residual Heat Removal, Chemical and Volume Control System), serve three primary functions, which include: removing stored and fission product decay heat, controlling reactivity, and precluding reactor vessel boron precipitation.

Response to Design Basis Events are typically defined by the source of water and flow path. The initial phase is called injection. The Residual Heat Removal System pumps would operate in recirculation unless the Reactor Coolant System pressure was less than the shut-off head of the Residual Heat Removal System pumps. If the Reactor Coolant

System pressure is less than shut-off head, Residual Heat Removal System would inject via the cold legs utilizing the Refueling Water Storage Tank as the suction source. When the Refueling Water Storage Tank reaches the low level setpoint, the Residual Heat Removal System pump suction is automatically swapped to the containment sump, which is called the recirculation phase. The Residual Heat Removal System provides suction to the Safety Injection System and Chemical and Volume Control System pumps, which supply water to the Reactor Coolant System cold or hot legs.

The ECCS also provides core cooling and prevents excessive boron concentration in the reactor vessel through hot leg recirculation at the top of the reactor core. ECCS system design and operation precludes boric acid concentrations in excess of the solubility limit which might otherwise reduce the overall heat transfer capability. ECCS systems can inject water into both the hot and cold legs. If a break is on a hot leg, the Chemical and Volume Control System pumps continue to inject water through the cold legs, which must pass through the core to the break. If the break is on a cold leg, the Safety Injection System pumps (and possibly the Residual Heat Removal System pumps) inject water through the hot legs, which pass through the core to exit at the break. The flow through the core reduces the boron concentration in the core area. Small break LOCAs may not require transfer to hot leg recirculation at all. Transfer to hot leg recirculation is a function of break size and location.

As described in UFSAR Section 6.3.2, if hot leg recirculation is necessary, only a small amount of hot leg flow is necessary to make up for boil-off to remove the decay heat at the swapover time and prevent exceeding the NRC mandated limit on the core region boron concentration. If at least this much flow can be provided by the available Safety Injection Pump when aligned for hot leg recirculation, a direct Residual Heat Removal System hot leg injection flow path need not be aligned. The peak pressure transient analysis for containment takes credit for a certain amount of Residual Heat Removal System Auxiliary Containment Spray. Because of the relative resistances of the two flow paths, adequate Residual Heat Removal System spray cannot be guaranteed if the only available Residual Heat Removal System Pump, assuming single failure, is aligned for both injection and spray. A single Safety Injection Pump, at the time of hot leg recirculation alignment in the emergency procedures, is predicted to be able to supply much more than the minimum amount of core cooling and boron dilution flow required. Therefore, the Residual Heat Removal System Pump need not be aligned for direct hot leg injection for accidents evaluated in the UFSAR. Per UFSAR section 6.3.2, Safety Analysis does credit this flow path as an ECCS safety function. In the Catawba SER, the NRC did not specifically require Residual Heat Removal System Pump injection to the hot legs as one of the pathways to prevent core boron precipitation and flow blockage. The UFSAR provides an updated time requirement and safety analysis that does not take credit for this Residual Heat Removal System hot leg recirculation path. Therefore, the open function testing for these valves will be moved to the Supplemental Testing program. The Supplemental program will continue to test these valves, however the testing requirements are not as rigorous as the OM-10 program.

Technical Specification 5.5.8 requires testing of pumps and valves per ASME Section XI plus applicable addenda as required by 10CFR Part 50, Section 50.55a except where specific written relief has been granted per 50.55a(g)(6)(i).

The existing code, OM Part 10, requires testing those valves which perform a specific function in shutting down a reactor to the cold shutdown condition, maintaining the reactor in the cold shutdown condition, or in mitigating the consequences of an accident.

The safety related Reactor Coolant System Pressure Isolation function, described in UFSAR Table 5-41, is met by performing procedure PT/1(2)/4200/01N. The closed function test is not affected by this change to remove IWV testing of the open function.

Technical Specification 5.5.8 requires testing in accordance with 10CFR50.55a. This is being done by testing the hot leg recirculation check valves from both Safety Injection pumps. Hot leg injection from the Residual Heat Removal System Pumps is not credited with any Residual Heat Removal System or Safety Injection safety function. Therefore no changes to the Technical Specification are required. Section 3.9.6 of the UFSAR requires testing in accordance with 10CFR50.55a, however specific details are not provided, therefore a change to the UFSAR will not be required.

Evaluation: This change does not involve an unreviewed safety question. The change does not change the facility as described in the UFSAR, nor does it change the test procedures, methods, or acceptance criteria of testing already being performed for these valves. Therefore the IST Program Manual change are not tests or experiments, nor are they significant enough to justify inclusion in the UFSAR. This IST Manual change does not result in procedure and/or hardware changes to the facility. There is no adverse effect on the overall performance of the systems involved nor will it cause any systems to be operated outside of their design limits. The likelihood of the accidents evaluated in the SAR occurring will not be increased as result of the described activity. Performance of systems assumed to function in the accident analysis will not be affected. Required testing is being performed. Therefore, the proposed change to the IWV program will not increase the probability of malfunction of any equipment important to safety.

The activity being evaluated will not result in an increase in dose from any accident or equipment malfunction. No changes, degradation or prevention of any actions described in the UFSAR will occur as a result of this activity. All applicable components are being tested as required per Technical Specification 5.5.8 and UFSAR Section 3.9.6, and this activity will not change, degrade or prevent any actions described in the UFSAR. The effect of these changes has been evaluated and judged to be acceptable since there is no adverse effect on Reactor Coolant System pressure boundary, containment, and system safety support functions. None of the assumptions previously made in evaluating the radiological consequences of an accident will be adversely affected.

No new credible failure modes of fission product barriers or accident mitigating equipment are created. Should a previously assumed malfunction of a syatem, structure or component important to safety occur, the activity does not result in increased radiological consequences. No common mode failures or invalidation of separation criteria occur as a result of the proposed activity. Therefore, there is no increase in consequences of such a malfunction of equipment important to safety evaluated in the UFSAR.

There are no new credible failure modes or operating characteristics as a result of the change being evaluated. This aspect of this activity will not create the possibility for an accident not considered in the UFSAR.

These changes cannot create malfunctions or new possibilities of malfunctions different than those evaluated in the UFSAR. All necessary and required (applicable) testing is being performed. Therefore, the activity cannot lead to an equipment failure mode of a different type from those analyzed in the UFSAR.

The margin of safety to any Technical Specification will not be decreased due to these changes. The safety limit and design parameters for these systems and individual components are not affected. These valves are being tested by approved procedures that will ensure operational readiness of the required safety functions in the event of an accident as required by Technical Specification 5.5.8 and UFSAR Section 3.9.6.

The open function for these valves will be removed from the OM-10 inservice testing. This function, though not required by OM-10, will be moved to a supplemental program which is being developed. Test type and frequency will be commensurate with safety significance. No Technical Specification changes are required. No UFSAR changes are required.

94 **Type:** Miscellaneous Items **Unit:** 0

Title: Catawba Nuclear Station Inservice Program Manual, Revision 25

Description: The Catawba Inservice Testing Program will be revised to add a Supplemental Testing Program. Currently Catawba does not perform supplemental testing. The Supplemental Program would include components not specifically required by the Section XI (OM-6, OM-10) Code, but could be covered by 10CFR50 Appendix B. At Catawba, the approach has been to perform ASME Section XI testing on all components which fit the criteria for inclusion (10CFR50.55a, ASME Section XI, OM6, OM-10) and evaluate other components for inclusion on a case-by-case basis. The philosophy for testing 'other components' has slowly been evolving to the point where a supplemental program is prudent. A supplemental program would test those components which are deemed important to safety but do not fit the ASME Section XI criteria.

The components, which are being included in the initial development of the supplemental program, are all valves. 10CFR50 Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, is the basis for the supplemental program. Discussion in the CFR suggests the prudence of testing components that are not explicitly covered by ASME testing, but are important to safety. Catawba's past philosophy has been to test components, which may not have been required by the Section XI Code but were important to plant operation, by Section XI rules. There were very few of these components. While not explicitly required, the need to test additional components covered by 10CFR50 Appendix B, is reasonable based on their safety significance. As more of these components are identified which fit the supplemental program guidelines, development of this program allows flexibility in defining testing requirements, acceptance criteria and frequencies. With the development of this program, components which were included previously in the 'formal' IST program but could be part of the supplemental program, are being moved to the supplemental program.

The Program Document contained in the Inservice Testing (IST) Manual currently has a section titled "Appendix B: 10CFR50, Appendix B Program (Supplemental Testing Program)" which provides guidelines for supplemental testing. This section was included as a generic enclosure of the Program Manual even though Catawba had no supplemental program. This section was 'standard issue' for Duke Power IST Programs. As this section already exists in the manual, this change will merely list those affected components with the testing requirements. Again, these components do not fit the requirements as defined in Section XI, parts OM-6 and OM-10. These components are important to safety and should be tested to insure operational readiness. Testing will be conducted per approved procedures. Existing frequencies and acceptance criteria will not change. New frequencies and acceptance criteria will be specified by the IST Engineer.

The relief valves being added to the supplemental program are those recommended by calculation CNC-1249.00-00-0060 (an Operability Evaluation) and Design Basis for Testing Duke Class A, B, C, E, and F Relief Valves. The relief valves being added are KC-313, KC-330, KC-355, KC-361, KC-374, KC-380, KC-386, KC-392, KC-404, KC-814, NB-331, NF-895, NI-481, RN-499, RN-807, RN-815, RN-823, RN-861, RN-863, WL-462, WL-826, and WL-A33. The logic for including these valves in the supplemental program is in the calculation. The calculation also references Design Study CNDS-113,

which evaluated these valves and determined testing requirements. Some testing requirements, as identified in the Design Study, were modified by the calculation. These changes were to move several relief valves from the supplemental program, as recommended by the study, to the IST program. These changes have been made. The only remaining piece is to add the balance of the valves to the supplemental program. Required testing is being performed as part of the relief valve testing program. In the relief valve testing program, a representative sample of the total population is tested each outage. The results of these tests determine if other relief valves need to be tested. Procedures are in place to handle this testing. The only change is to add these valves to the supplemental program,

Several Auxiliary Feedwater System check valves are being included due to their use during a steam generator overfill accident . These valves are nuclear safety related on Unit 1 and non-nuclear safety related on Unit 2. The Unit 1 valves will eventually be reclassified as non-nuclear safety related. Because the steam generator overfill accident: was not part of the original licensing basis, the NRC has allowed non-safety components to be used. Since these valves will be non-nuclear safety related, they are excluded from the code required testing. However, because they are required to close to isolate the Auxiliary Feedwater System Control Valve air accumulators from a loss of Instrument Air (accumulators provide air to Auxiliary Feedwater control valves), they will be included in the supplemental program. These valves are CACK360, CACK361, CACK480, CACK481, CACK520, CACK521, CACK640, AND CACK641. A Procedure entitled, "Auxiliary Feedwater Flow Control Valve Air Accumulator Leakage Test", verifies that these check valves close. Pressure gauges are observed after instrument air is isolated and verifies piping downstream of the subject check valves remains pressurized. Acceptance criteria is based on pressure required to cycle the Auxiliary Feedwater Control Valves for the steam generator overfill accident. This testing is currently being performed on an 18 month frequency.

Also, the open function for Safety Injection System valves NI-15 and NI-129 is being moved into the supplemental program. The "Closed to Open" stroke was previously in the IST program, however it was determined that this stroke did not meet the intent of the program. This was evaluated under separate 10CFR50.59 evaluation. The "Closed to Open" stroke verifies a path from the Residual Heat Removal System to the Reactor Coolant System hot legs. While this function is not "safety related", it is a path which is credited in the safety analysis and will continue to be tested in the supplemental program. A Procedure entitled "NI Check Valve Stroke to B&C Hot Legs Full Stroke Test", performs the "Closed to Open" test on a Cold Shutdown frequency. The "Open to Closed" stroke will remain in the IST program as these valves are Reactor Coolant System pressure isolation valves and must close.

Technical Specification 5.5.8 requires testing of pumps and valves per ASME Section XI plus applicable addenda as required by 10CFR Part 50, Section 50.55a except where specific written relief has been granted per 50.55a(g)(6)(i).

The existing code, OM-6 & OM Part 10, requires testing those components which perform a specific function in shutting down a reactor to the cold shutdown condition, maintaining the reactor in the cold shutdown condition, or in mitigating the consequences of an accident.

10CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants", provides guidance for design, fabrication, construction and testing of structures, systems and components at these facilities.

Evaluation: This change to the IST Program does not involve any unreviewed safety questions. These changes do not create any new failure modes or operating characteristics. All required testing is being performed by approved programs and procedures. No changes to the Technical Specifications are required. No UFSAR changes are required.

169 **Type:** Miscellaneous Items **Unit:** 0

Title: Catawba Unit 1 Fuel Cycle 13 Reload Safety Evaluation

Description: (C1C13) core reload in calculation file CNC-1552.08-00-0315. The effect of any other plant changes made concurrent with the refueling outage are not addressed in the calculation.

The C1C13 Reload Design Safety Analysis Review (REDSAR), performed in accordance with Nuclear Engineering Division workplace procedure NE-102, " Workplace Procedure for Nuclear Fuel Management", serves as the unreviewed safety question evaluation. The SAPP section of the REDSAR checklist indicates the need to further evaluate the BOC minimum prompt neutron lifetime, the BOC minimum fuel temperature, the power shapes for the steam line break event and the dropped rod event, and the total peaking at Beginning of Cycle Hot Zero Power for the rod ejection event. These evaluations, documented in CNC-1552.08-00-0315, confirm that the updated safety analysis report (UFSAR) Chapter 15 accident analyses remain bounding with respect to the C1C13 safety analysis reactor physics parameters. The safety analysis reactor physics parameters method is described in Topical Report DPC-NE-3001-PA, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology." The transition core analysis for the RFA fuel type are performed according to DPC-NE-2009-PA, "Westinghouse Fuel Transition Report," and WCAP-12945-PA, "Best Estimate Analysis of the Large Break Loss of Coolant Accident for the McGuire and Catawba Nuclear Stations."

Evaluation: The C1C13 core reload is similar to past cycle core design, with a design generated using NRC approved methods. No Technical Specification changes specifically related to the operation of the C1C13 core are required. There are no unreviewed safety questions associated with the C1C13 core reload. No UFSAR changes are required.

44 Type: Miscellaneous Items **Unit:** 0

Title: Compensatory Action for PIP C-98-1726, Addendum to Rev 2

Description: Catawba Corrective Action Program Problem Report C99-2572 was written in June 1999 to address a discrepancy concerning the evaluation of the UFSAR Section 15.2.2 "Loss of External Load Transient" in the 10CFR50.59 evaluation for compensatory actions placed on the Auxiliary Feedwater System. The compensatory actions involved closing valve 1(2)CM-127 to prevent an adverse system interaction which could cause Upper Surge Tank water to become hotter than the maximum temperature allowed for suction sources to the Auxiliary Feedwater System. When the original operability evaluation was done for this problem in 1998, the 10CFR50.59 evaluation noted that all Loss of External Load Transients had always resulted in a reactor trip. Therefore the effects of this transient were not evaluated in the original 10CFR50.59 evaluation. Since that time there has been a Loss of External Load Transient on Unit 2 that was successfully mitigated without a reactor trip. Therefore the 10CFR50.59 evaluation was revised to reevaluate whether isolating valve 1CM-127 with regard to the Loss of Load Transient of UFSAR Section 15.2.2 is acceptable per 10CFR50.59.

Evaluation: The statement in Revision 0 and Revision 1 of the 10CFR50.59 evaluation read as follows concerning UFSAR Section 15.2.2:

> "Isolation of CM-127 will not cause or increase the probability of a loss of external load. In this analysis the reactor does not trip. However, testing of the full load rejection always resulted in a turbine trip due to overspeed which therefore results in a reactor trip."

The original evaluation of this transient based on past history concluded that the Condition I Loss of Load Transient resulted in a Condition II Turbine Trip and therefore a Loss of Feedwater, did not need to be considered since it was also a Condition II tranisent. Since the successful Runback of Unit 2 on Loss of External Load, the conclusions of the previous evaluations concerning the Loss of External Load are no longer true. Since the successful Unit 2 Runback, it is evident that the compensatory actions to isolate valve CM-127 did not cause a Condition II Loss of Feedwater and Turbine Trip Transient during that particular transient. Furthermore, it can be also concluded that the Loss of External Load Transient does not increase the probability of Loss of Feedwater or Turbine Trip since the Heater Drain Pumps automatically trip on a full load rejection signal. Because the Heater Drain Pumps receive an automatic trip signal in this transient, the Condensate Booster Pumps will not trip on low flow due to continued operation of the Heater Drain Pumps. The original evaluation establishes that the Main Feedwater Pump recirculation valves have sufficient capacity to provide minimum flow to the Condensate Booster Pumps as long as the Heater Drain Pumps are not in operation. Therefore, operation of the Condensate and Feedwater System is not affected by the requirements to mainatain valve CM-127 isolated in the Loss of Load Transient. There are no unreviewed sfaety questions associated with this Compensatory Action. No Technical Specification changes are required. No UFSAR changes are required.

168 **Type:** Miscellaneous Items **Unit:** 0

Title: Compensatory Action for Reactor Building Penetration Disassembly

Description: The Compensatory Action for Reactor Building Penetration Disassembly is used to ensure that the design functions provided by the Reactor Building wall and its associated penetrations will be maintained if one or more of the penetrations is disassembled or open in Modes 5, 6 or No Mode. During outages it may become necessary to disassemble various flanged equipment penetrations in the Reactor Building wall to provide temporary access for cables, hoses, etc. Having these penetrations open is significant because when a Unit is in Modes 5, 6, or No Mode the Reactor Building wall between the Auxiliary Building and the annulus becomes a Tornado Pressure Boundary for the Auxiliary Building. This occurs because (1) the boot seal on the shutdown unit, is removed exposing the annulus to negative pressures that could be induced by a tornado, (2) the equipment hatch shield door is rolled back thus removing tornado missile protection for the containment vessel, and in particular the equipment hatch, for the shutdown unit, and (3) typically, the airlock doors on the shutdown unit are maintained open to minimize wear on the door seals. Additionally, other openings, such as the personnel airlocks, annulus doors, Containment Purge Ventilation System ductwork and Annulus Ventilation System ductwork, may be opened which would allow for air to pass from the Auxiliary Building to the Reactor Building/containment. Besides tornado protection the Reactor Building wall and penetrations provide Fire Barrier and Security Barrier functions. Following the guidance given in this compensatory action ensures that all design functions will be provided or can be restored in sufficient time to protect the Auxiliary Building.

Evaluation: This Compensatory Action does not involve an unreviewed safety question. No Technical Specification changes are required. No UFSAR changes are required.

219 **Type:** Miscellaneous Items **Unit:** 0

Title: Compensatory Action for Residual Heat Removal/Containment Spray Sump Pump Interlocks per PIP C98-4098, Revision 2

Description: Corrective Action Program Report (PIP) Serial Number C98-4098 identified the fact that an interlock between the SSPS and the Residual Heat Removal/Containment Spray Sump Pump controls, described in UFSAR Section 6.3.2.5, was never installed at Catawba. The purpose of the interlock was to secure the the Residual Heat Removal/Containment Spray sump pumps upon a safety injection such that the sump pumps could not run in "Auto" mode or at the "high" setpoint of the sump level switches. There is no control room alarm for sump "high" level. Securing the sump pumps in "auto" as a result of a safety injection would ensure that any post-LOCA ECCS system leakage to the sump would result in the sump level reaching the "high-high" setpoint. At the "high-high" setpoint, the sump pumps would activate and a control room computer alarm would be generated. Because the capacity of one Residual Heat Removal/ Containment Spray sump pump is greater than the maximum postulated leak rate of a failed ECCS pump seal (50 gpm), the interlock was intended to provide assurance that the control room would be aware of post-LOCA leakage to the the Residual Heat Removal/Containment Spray sump.

As a result of the missing interlock, a compensatory action was initiated to maintain all of the Residual Heat Removal/Containment Spray sump pumps in "standby" mode. While in "standby", the sump pumps will not activate until sump level reaches the "high-high" setpoint. Since a control room alarm is generated at the "high-high" sump level setpoint, the compensatory action mimics the intent of the missing interlock. However, the original compensatory action did not make any allowances for temporarily placing the the Residual Heat Removal/Containment Spray sump pumps in "auto" mode to support modification or other work.

Revision 2 to the compensatory action provides additional instructions for temporarily placing the sumps pumps in "auto" mode to support modification or other work. The compensatory action revision was prepared specifically to support implementation and testing of modification CN-21405, which will install the missing interlock on Unit 2. Specific conditions must be met for the sumps pumps to be placed in "auto" while the compensatory action is in effect. Any time the Residual Heat Removal/Containment Spray Sump Pumps are placed in "auto", a qualified person must be stationed at the sump pump control panel, 1ELCP0243, located on elevation 543' of the Auxiliary Building, In the event of a safety injection, all the Residual Heat Removal/Containment Spray Sump Pumps must be immediately (as soon as possible) switched to the "standby" position. Adequate communication must be assured between the control room and the person stationed at the panel (plant page, radio, pager, etc.). Upon completion of work, the sump pump controls must be returned to the "standby" position.

Evaluation: The Catawba Environmental Qualification Criteria Manual (EQCM) (CNLT-1780-03.03) describes the environmental conditions expected during normal and postulated accident conditions for radiation, pipe rupture and HVAC criteria. One purpose of this safety review is to confirm that post accident environmental conditions at the the Residual Heat Removal/ Containment Spray Sump Pump local control panel will allow the compensatory actions described above to be performed. Because the Residual Heat Removal/Containment Spray Sump Pump local control panel is located in the main area

of the 543' elevation of the Auxiliary Building, there are no normal operation environmental conditions that would prevent performance of the compensatory action (switching sump pump controls from "auto" to "standby" upon notification of a safety injection). Also, the post-LOCA radiation dose rate in the Auxiliary Building zone that contains the Residual Heat Removal/Containment Spray Sump Pump control panel would be negligible and thus would not preclude performance of the compensatory action. Also, the Environmental Qualification analysis for HVAC temperatures does not indicate that post accident temperatures in the Residual Heat Removal/Containment Spray Sump pump control panel zone would prevent personnel from performing the compensatory action. The analysis for HVAC temperatures is separate from the analysis for pipe rupture described below.

Per the pipe rupture analysis, the zone that contains the the Residual Heat Removal/Containment Spray Sump Pump control panel could encounter a high temperature steam environment and water spray with an ambient temperatures of 238 degrees F. and a steam spray temperature of 275 degrees F. These conditions would be the result of a rupture of an Auxiliary Steam System pipe that runs in the overhead near the sump pump control panel. Obviously, these extreme temperature conditions would not allow an operator to perform the compensatory action of switching the Residual Heat Removal/Containment Spray Sump pumps from "auto" to "standby" upon notification of a safety injection. However, the ultimate purpose of the compensatory action and the missing interlock is to provide a method for the control room to be made aware of post LOCA ECCS System leakage to the Residual Heat Removal/Containment Spray sump. Thus, using the Single Failure Criterion of CNS-1465.00-00-0001, Rev. 3; a steam pipe rupture does not have to be assumed to occur concurrently with a primary system LOCA. Pipe ruptures are generally hypothesized to be caused by seismic events. Seismic events are not assumed to occur concurrently with a LOCA. Thus, the analysis for pipe rupture does not apply to the scenario of the compensatory action and environmental conditions would not prevent an operator from accomplishing the compensatory action.

Revision 2 of the compensatory action will not prevent the Residual Heat Removal/Containment Spray Sump pumps from performing the safety related function, described in UFSAR Section 11.2.2.2.4.3, of protecting safety equipment from flooding. In addition, the compensatory action will continue to mimic the function of the missing interlock described in UFSAR Section 6.3.2.5, by ensuring that the sump pump controls are placed in "standby" mode in the event of a safety injection.

There are no unreviewed safety questions associated with this compensatory action revision. The Residual Heat Removal/Containment Spray Sump Pumps are not accident initiators as described in Chapter 15 of the UFSAR and thus the probability of occurrence of an accident cannot increase as a result of the revision to the compensatory action. The compensatory action revision does not increase the probability of a malfunction of the Residual Heat Removal/Containment Spray Sump Pumps or the ECCS Pumps. The compensatory action revision simply adds an allowance for temporarily placing the sump pumps in "auto" to accomplish work and provides instructions for returning the sump pumps to "standby" upon completion of the work or upon notification of a safety injection. The sump pumps are designed to operate in either "auto" or "standby" mode so there will be no detrimental effect to the sump pumps as a result of the compensatory action revision. The Residual Heat Removal/Containment Spray Sump Pumps do not

mitigate the consequences of an accident as described in Chapter 15 of the UFSAR. The compensatory action revision will not affect the ability of the Residual Heat Removal/Containment Spray Sump Pumps to pump down the 522' elevation of the Auxiliary Building and the function of protecting other safety related equipment from flooding will not be affected. The compensatory action will continue to assure that the control room has a method to identify post-LOCA, ECCS System leakage into the Residual Heat Removal/Containment Spray Sump. The compensatory action revision will not change the designed operation of the Residual Heat Removal/Containment Spray Sump Pumps or any other safety related system or component. Therefore, increasing the consequences of a malfunction of this equipment will not be possible. No Technical Specification changes are required. No UFSAR changes are required.

111 **Type:** Miscellaneous Items **Unit:** 0

Title: Compensatory Action Guidelines for Firestop Penetration AX-635-W-028 for work associated with W/O 97049829

Description: Compensatory actions are being established for breaching firestop penetration number AX-635-W-028. This penetration is a Selected Licensing Commitment Fire Barrier and a UFSAR committed Tornado Pressure Boundary. The work involved is maintenance work per work order 97049829 to clean tank 1WLTKO039. A two inch diameter stainless steel pipe will be installed and removed in the firestop penetration, and the fire boundary will be established again by the pipe that extends five feet on either side of the faces of the wall and the firestop material that is placed around the newly installed pipe. This arrangement is satisfactory for a fire penetration seal based on Duke Power Fire Protection Specification DPS-1435.00-00-002. The tornado pressure boundary will be established again by the use of pipe caps on the ends of the stainless steel pipe that will easily resist the postulated three psi tornado depressurization that only occurs for a period of three seconds. When the pipe caps are not in place, non collapsible reinforced hose will be fastened to the ends of the stainless steel pipe. Either a pipe cap or a hose must be installed on one of the two ends of the pipe at all times. The hose is rated for 225 psi internal pressure and can easily resist the three psi tornado external depressurization. After the work has been completed on cleaning 1WLTKO039, the two inch diameter stainless steel pipe will be removed, and the firestop penetration and fire boundary will be repaired or replaced. The same compensatory action will be in place for the repair or replacement.

There are no ventilation or security concerns since the Auxiliary Building Ventilation system is on both sides of the wall, and the security requirements are the same on both sides of the wall. There is no change related to tornado missile issues. There is no environmental qualification concern created. There is no C02 fire protection boundary involved.

The compensatory actions required here cover only the period while the stainless steel pipe is being installed through the partially removed fire penetration or the period while the pipe is being removed and the fire boundary reestablished.

The compensatory actions consist of the following:

A Fire Watch is required to be put in place any time either of the blind flanges are removed or any time the fire boundary is not completely in place.

Provisions must be made to reinstall the fire penetration and at least one pipe cap (for the situation where the pipe is left in the penetration). Alternately the fire boundary seal may be installed completely (without the pipe in place). Whichever of these two actions is taken must be complete within one hour of a Tornado Watch or Warning in York County, South Carolina. This one hour must take into account the half hour curing time of the material used in the fire boundary or seal.

Evaluation: The reliability of the fire barrier will be maintained by the performance of the firewatch. This approach to ensuring fire barrier reliability does not increase the probability of an accident evaluated in the UFSAR. It is noted that the penetration will be "closed" within

an hour of a Tornado Watch or Warning (even if the installation job were to take longer than one hour), thus ensuring the integrity of the tornado pressure boundary. Therefore, this compensatory measure will not increase the probability of an accident evaluated in the UFSAR. There are no unreviewed safety questions associated with this compensatory action. No Technical Specification changes are required. No UFSAR changes are required.

152 **Type:** Miscellaneous Items **Unit:** 1

Title: Compensatory Action Guidelines for Nuclear Service Water System Pipe Cleaning

Description: During the 1EOC12 Refueling Outage, Nuclear Service Water System supply piping will be cleaned from the outlet of the Nuclear Service Water System Pump Strainers to the Component Cooling, Diesel Jacket Water Cooling and Containment Spray Heat Exchanger inlets on both Trains. This cleaning effort will require opening manways M-6-1, M-8-1 and the 30 inch buried piping leading into the Auxiliary Building. Compensatory Action Guidelines will be issued for this pipe cleaning activity. Access to the pipe will be gained through the flange openings by removing valves/equipment and manways. This will result in direct access to the Auxiliary Building from outside the protected area. The Compensatory Action Guidelines for Nuclear Service Water System Pipe Cleaning will provide plant personnel with information necessary to ensure the plant stays within its design basis during the activity. Several concerns associated with opening this Nuclear Service Water System piping are addressed below.

Tornado Pressure Barrier

Cleaning of Nuclear Service Water System piping requires opening the 30 inch buried piping leading into the Auxiliary Building. A compensatory action is required to protect nuclear safety related equipment within the Auxiliary Building from the negative pressure that would be created if a design basis tornado passed over an open manway while the 30 inch piping is opened in the Auxiliary Building. A design basis tornado event results in a 3 psi differential pressure between the Auxiliary Building and outside.

A compensatory action for an open pathway into the Auxiliary Building during a tornado requires that provisions be specifically provided to allow closure of the opening within one hour of the establishment of Tornado Watch or Warning conditions in York County. A metal plate has been designed to cover the 30 inch pipe opening in the event of a tornado. To facilitate installation of the cover plate at least two holes will be drilled to match existing flange stud locations. The two studs are sufficient to prevent the plate from falling off during a seismic event.

Flooding

Cleaning of Nuclear Service Water System piping during 1EOC12 requires opening the 30 inch buried piping leading into the Auxiliary Building. Open Manway M-8-1 is located at elevation 597'-3" and open manway M-6-1 is located at elevation 588'-6". A compensatory action is required to protect nuclear safety related equipment in the Auxiliary Building from the effects of water flooding into the open manways. Compensatory Action Guidelines for Nuclear Service Water System Pipe Cleaning Flood levels for the Catawba site are analyzed for the following flood producing phenomena:

1. Probable Maximum Flood (PMF) resulting from the Probable Maximum Precipitation (PMP) in the drainage area.

2. A 25 year frequency flood passing through Lake Wylie combined with a seismic failure of Cowans Ford Dam, the largest upstream reservoir.

3.A Standard Project Flood (SPF) passing through Lake Wylie combined with the failure of one of the upstream dams due to an Operating Basis Earthquake (OBE). The SPF is considered equal to one-half of the PMF.

A compensatory action for an open pathway into the Auxiliary Building during a flood requires that provisions be specifically provided to allow closure of the opening in the event of a Design Basis Flood. The plate designed to close the 30 inch open Nuclear Service Water System pipe during a tornado warning will be installed to minimize the amount of water entering the Auxiliary Building during a design basis flood. Leakage around the plate will flow into the buildings floor drain system.

During a design basis flood, water will leak into the Auxiliary Building around cracks at each outside door. Water will also leak into the Auxiliary Building around the plate covering the 30 inch Nuclear Service Water System pipe. As water enters the building it spreads across the floor and is intercepted by the floor drain system. The floor drain system routes the entire volume of water to four floor drain sumps and a floor drain tank, all located in the Auxiliary Building at elevation 543+0. Assuming no pumping from the sumps, water will pond as follows:

Unit 1 and 2 Aux Feedwater Pump Room and Shutdown Panel Room (Elevation 543' - 0") - 4 inches
Unit 1 and 2 Outside Doghouse (Elevation 577' - 0") - 3 inches
Unit 1 and 2 UHI Building (Elevation 550'+0") - 4 inches
Floor Drain Tank Room (Elevation 543' - 0") - 2 inches
Waste Evaporator Package Room (Elevation 537' - 0") - 3 feet, 3 inches
Chemical Drain Tank and Pump Room (Elevation 537' - 0") - 6 feet (completely filled)

Using the PMP values and time distributions in the UFSAR, no safety related equipment will be affected by the resulting water levels from the PMP event, and the plant can be safely shutdown by established normal shutdown procedures.

A flood inflow hydrograph developed for PMP occurring locally over the Big Allison Creek drainage area is also considered to coincide with Standard Project Flood over the Lake Wylie drainage area along with a seismic failure of Mountain Island Dam. Such a highly unlikely combination of events, causes a flood elevation of 578.1 ft in Lake Wylie, lower than the surcharge caused by PMF.

An analysis has been performed to evaluate the failure of Cowan's Ford Dam and the SPF to find the highest flood level. The critical flood hydrograph associated with an upstream dam breach was developed for the flood resulting from the PMP occurring locally over the Big Allison Creek drainage area, combined with the SPF over the Lake Wylie drainage area and a seismic failure of Mountain Island Dam. This inflow will cause a flood elevation of 578.1 feet in Lake Wylie, which is lower than the surcharge caused by the PMP on the larger Lake Wylie drainage area. Since the flood elevation is lower than the lowest manway (M-6-1) being opened during the pipe cleaning, flood related events due to dam breaks or flooding will not require special action under these compensatory actions.Therefore, the only "flood" related issues are associated with a significant amount of rainfall.

Security

Security will use M-8-1 as the Security boundary during the entire Nuclear Service Water System pipe cleaning evolution. The placement of Security measures at this location will allow manway removal and entry for all associated Nuclear Service Water System piping within the Protected Area and the Owner Controlled Area.

Security will begin patrolling M-8-1, M-9-1 and the Nuclear Service Water System Pump Structure upon notification that the system is being dewatered.

M-8-1 will be the first manway outside of the Protected Area to be opened. A Security Officer will be posted at M-8-1 prior to loosening and opening the manway. Additional manways are to remain secure until Security accepts the barrier being installed at this location. Personnel entering the manway to clean the pipe and install a double layered Security barrier within the pipe shall receive a pat down search to ensure no firearms, explosives or incendiary devices are placed in the piping that could be accessed from the protected side of the pipe.

Security shall inspect the inner and outer sections of the double layered barriers as it is being installed. Upon acceptance of the installation, Security shall place a shielded Security padlock that will secure access to the barrier. The key for the lock will be placed in the Security Secondary Alarm Station. Total duration is expected to be four to six hours for installation and securing the barrier.

Security will start patrolling the barrier installed at M-8-1 and discontinue patrols of M-9-1 and the Nuclear Service Water System Pump Structure. All other manways and access points to the Nuclear Service Water System may be opened at this point.

There may be a need to temporarily remove the barrier at M-8-1. Notification that the barrier needs to be removed shall be coordinated with the Nuclear Service Water System project team. Security compensatory measures must be established prior to the removal of the barrier. Maintenance personnel shall be searched prior to entering the manway to remove the barrier. Security will maintain positive control of all access into the manway and prevent any unauthorized entry through the Nuclear Service Water System piping during anytime the barrier is removed. Securing the barrier shall be accomplished in the same method described above.

M-8-1 will be the last manway to be installed outside the Protected Area. Security will resume patrol of M-8-1, M-9-1, and the Nuclear Service Water System Pump Structure until the Nuclear Service Water System has been refilled.

Effects on Ventilation Systems

Cleaning of Nuclear Service Water System piping during 1EOC12 requires opening the 30 inch buried piping leading into the Auxiliary Building. The Auxiliary Building Ventilation System system normally keeps the Auxiliary Building at a slight negative pressure. With the 30 inch Nuclear Service Water System pipe opened the Auxiliary

Building pressure may increase a small amount. This will not affect the ECCS pump room pressure. The opened Nuclear Service Water System pipe will also tend to help the Control Room pressure. The Fuel Building will have an insignificant change in pressure due to the opened Nuclear Service Water System pipe. No significant changes will occur and no other ventilation systems will be affected.

The requirements of this Compensatory Action were reviewed as required for an infrequently performed evolution. The scope of this maintenance activity will require direct upper management involvement. This project is considered a "Critical Maintenance Process".

The actions allowed in this Compensatory Action have been evaluated against each other for potential adverse cumulative effects. In addition, this Compensatory Action has been reviewed against other Compensatory Actions and normal plant evolutions to identify potential cumulative effects. The results of these reviews indicate that adequate instructions are present to prevent any adverse cumulative effects. This review made the assumption that Operations would limit the number of open Compensatory Actions to a level that notifications could be made within the time constraints of the open Compensatory Actions.

Evaluation: There are no unreviewed safety questions associated with this Compensatory Action. The activity of cleaning the Nuclear Service Water System piping utilizing this Compensatory Action will not affect the probability or consequences of accidents analyzed in the UFSAR because cleaning the Nuclear Service Water System piping is not an accident initiator and the compensatory action guidelines ensure adequate protection of Auxiliary Building equipment. No changes to the Technical Specifications are required. No UFSAR changes are required.

156 **Type:** Miscellaneous Items **Unit:** 0

Title: Compensatry Action Guidelines - Plant Access Doors Revision 20

Description: The Compensatory Action -Guidelines Plant Access Doors serves as the overall review of doors within the Auxiliary Building (generally the doors between the AA and QQ walls). These guidelines identify the design features of each of the reviewed doors. The guidelines allow activities which could prevent the doors from closing in their normal manner to occur as long as they do not challenge a safety-related function or if the feature they are designed to protect can be removed from service and declared inoperable. The guidelines allow doors to be restricted from closing with certain compensatory actions. These compensatory actions are limited to Fire Boundary Doors which require a Fire Watch, Tornado Doors which must be capable of being closed within one hour, and Security Doors which require Security Access Control to be established. With the requirements of the Compensatory Action Guidelines Plant Access Doors satisfied, no Unreviewed Safety Questions are created and the margin to any Technical Specification is not challenged or reduced.

Revision 20 of the Compensatory Action Guidelines was initiated to incorporate changes associated with two station problem reports. Prior to this revision, several doors within the Auxiliary Building, Fuel Building and the Nuclear Service Water Pump Structure had not been evaluated and were not included in the guidelines. These station problem reports addressed ensuring all doors within these areas were evaluated and listed in the Compensatory Action Guidelines. This revision added ninety-six new doors to the compensatory action guidelines and expanded the scope of the Compensatory Action to include Fuel Building and Nuclear Service Water Pump Structure doors. Another change was to revise the Environmental Qualification zones associated with door AX221. The zone on one side of this door changed as a result of minor modification CE-61629. Two doors (AX314 #3 and AX656) were deleted from the Compensatory Action since they no longer exist in the plant. None of these changes alter the intent of the original 10CFR50.59 evaluation or the original compensatory action.

Evaluation: This revision to the Compensatory Action Guidelines will not effect the probability or consequences of accidents described in the UFSAR. There are no unreviewed safety questions associated with this change to the Compensatory Action Guidelines. No Technical Specification changes are required. No UFSAR changes are required.

157 Type: Miscellaneous Items **Unit:** 0

Title: Compensatry Action Guidelines - Plant Access Doors Revision 21

Description: The Compensatory Action Guidelines - Plant Access Doors serves as the overall review of doors within the Auxiliary Building (generally the doors between the AA and QQ walls). These guidelines identify the design features of each of the reviewed doors. The guidelines allow activities which could prevent the doors from closing in their normal manner to occur as long as they do not challenge a safety-related function or if the feature they are designed to protect can be removed from service and declared inoperable. The guidelines allow doors to be restricted from closing with certain compensatory actions. These compensatory actions are limited to Fire Boundary Doors which require a Fire Watch, Tornado Doors which must be capable of being closed within one hour, and Security Doors which require Security Access Control to be established. With the requirements of the Compensatory Action Guidelines Plant Access Doors satisfied, no Unreviewed Safety Questions are created and the margin to any Technical Specification is not challenged or reduced.

Revision 21 of the Compensatory Action Guidelines was initiated to incorporate changes associated with Technical Specification Amendments No. 187 (Unit 1) and No. 180 (Unit 2). These amendments established actions to be taken for an inoperable ventilation system due to a degraded Control Room pressure boundary or ECCS pressure boundary. Along with these changes other minor, editorial changes were made. None of these changes alter the intent of the original 10CFR50.59 evaluation or the original compensatory action.

Evaluation: This revision to the Compensatory Action Guidelines will not effect the probability or consequences of accidents described in the UFSAR. There are no unreviewed safety questions associated with this change to the Compensatory Action Guidelines. No Technical Specification changes are required. No UFSAR changes are required.

193 **Type:** Miscellaneous Items **Unit:** 0

Title: Compensatry Action Guidelines - Plant Access Doors Revision 23

Description: The Compensatory Action - Guidelines Plant Access Doors serves as the overall review of doors within the Auxiliary Building, Fuel Building, and Nuclear Service Water Pump Structure. These guidelines identify the design features of each of the reviewed doors. The guidelines allow activities which could prevent the doors from closing in their normal manner to occur as long as they do not challenge a nuclear safety-related function or if the feature they are designed to protect can be removed from service and declared inoperable. The guidelines allow doors to be restricted from closing with certain compensatory actions. These compensatory actions are limited to Fire Boundary Doors which require a Fire Watch, Tornado Doors which must be capable of being closed within one hour, and Security Doors which require Security Access Control to be established, and certain pressure boundary doors which are allowed to be opened under Technical Specification actions. With the requirements of the Compensatory Action Guidelines Plant Access Doors satisfied, no Unreviewed Safety Questions are created and the margin to any Technical Specification is not challenged or reduced.

Revision 23 of the Compensatory Action Guidelines was initiated to incorporate changes associated with Doors AX214A and AX214B. These doors were incorrectly listed in Revision 21 as part of the ECCS Pump Room pressure boundary. This Revision (Revision 23) is made to correct this error. The changes made to the compensatory action are considered editorial.

Evaluation: This revision to the Compensatory Action Guidelines will not effect the probability or consequences of accidents evaluated in the UFSAR. There are no unreviewed safety questions associated with this change to the Compensatory Action Guidelines. No Technical Specification changes are required. No UFSAR changes are required.

151 **Type:** Miscellaneous Items **Unit:** 2

Title: Revision of DC Channel Battery Service Test Requirements in Technical Specification Bases Section B 3.8.4 for Surveillance Requirement 3.8.4.8

Description: This change revises the DC Channel Battery (125 VDC Vital I and C Batteries EBA, EBB, EBC, and EBD) service test requirements in the Technical Specification Bases Section for SR 3.8.4.8. The change revises the discharge current values, the corresponding discharge times, and the required terminal voltage value to be consistent with both the 125 VDC Vital Instrumentation and Control Battery Duty Cycle shown in UFSAR Figure 8-25. (Note: the load values shown in Figure 8-25 include a 1.25 aging factor; hence, they are higher than the values in the Technical Specification Bases) and 125 VDC Vital Instrumentation and Control Battery and Battery Charger Sizing Calculation, CNC-1381.05-0011. In addition, the change will make the service test requirements for all four DC channel batteries consistent. The current service test requirements in the Technical Specification Bases still reflect the original battery configuration and battery sizing calculations. In the original battery configuration, batteries EBA and EBD were larger than batteries EBB and EBC. When the batteries were replaced in 1996 per modifications CN-11339 and CN-21339, batteries EBA and EBD were upsized from 1200 Ah to 1495 Ah and, batteries EBB and EBC were upsized from 825 Ah to 1495 Ah. To take advantage of the additional battery capacity, the battery sizing calculation was revised and a new, more conservative set of service test requirements for all four batteries were developed. Consequently, the new DC Channel battery service test requirements provide an additional margin of safety.

Evaluation: There are no unreviewed safety questions associated with this UFSAR change. No changes to the Technical Specifications are required. No UFSAR changes are required.

230 **Type:** Miscellaneous Items **Unit:** 0

Title: Revision to the Bases of Technical Specification 3.1.7 and Selected Licensee Commitment 16.7-11

Description: The Bases of Technical Specification 3.1.7 and Selected Licensee Commitment 16.7-11 are being enhanced to state that the Operator Aid Computer (OAC) Digital Rod Position Indication (DRPI) indication is a valid indication of actual control rod position. The following information will be added to the bases.

Gray code (A and B data from the data cabinets in containment) is sent to the DRPI equipment in the control room. The gray code is processed by the DRPI equipment and the rod position is displayed on the control board. The gray code is also sent from the DRPI equipment to the OAC, where it is processed by the OAC and the rod position is displayed on the OAC. The processing of the gray code by the DRPI equipment and the OAC are completely independent. Therefore, both the DRPI display and the OAC DRPI indication are considered valid indications of control rod position.

Evaluation: There are no Unreviewed Safety Questions associated with this change. No Technical Specification changes are required. No UFSAR changes are required.

61 Type: Miscellaneous Items **Unit:** 0

Title: SDQA Plan for SMARGINS, Version 7

Description: This change involves an upgrade to the Super-MARGINS (SMARGINS) software. The current software (SMARGINS - Version 6) is being replaced by Version 7. This software and the workstation it runs on are not a part of any System, Structure, or Component described in the UFSAR.

Evaluation: SMARG07 is an improved version of SMARG06. The methodology of calculating margin to thermal limits for LOCA FQ, DNB, and CFM are not changed. The new software incorporates modifications to support analyses outlined in DPC-NE-2009A for McGuire and Catawba Nuclear Stations, to continually reduce the Oconee Nuclear Station LOCA limit based on axial power shapes, and other minor enhancements. Part of this modification for DPC-NE-2009PA is to calculate the penalty factor used in the McGuire and Catawba Nuclear Station Improved Technical Specification Surveillance Requirements 3.2.1.3 and 3.2.2.2.

SMARG07 was certified by Duke Power's directive for software certification and verified to yield the same results as SMARG06, excepting the new modifications. The modifications were verified and are in compliance with Technical Specifications and approved methods.

The change involves no material change to the plant. The SMARGINS software and resident work stations are not a part of any system, structure, or component important to nuclear safety and do not directly affect any system, structure, or component. The SMARGINS software is not installed at the plant, but rather on workstations in the Nuclear General Office. A safety significant function performed by this system is to generate data to evaluate Improved Technical Specifications 3.2.1 and 3.2.2. The new software produces the same analytical results as the replaced software. Per DPC-NE-2009PA, the new penalty factors used in Technical Specification Surveillance Requirements 3.2.1.3 and 3.2.2.2 will accurately reflect the expected change in margin over the surveillance window and can be greater than or equal to the historic value of 1.02.

The assurance of fuel integrity limits associated with the referenced Technical Specifications are not compromised. This change does not impact any plant parameters, safety limits or setpoints that potentially affect the fission product barriers.

There are no unreviewed safety questions associated with this change. No Technical Specification changes are required. No UFSAR changes are required.

165 **Type:** Miscellaneous Items **Unit:** 0

Title: SDQA Plan for SMARGINS, Version 8

Description: This change involves an upgrade to the Super-MARGINS (SMARGINS) software. The current software, SMARGINS version 7, is being replaced by version 8. This software and the workstations it runs on are not part of any system, structure, or component described in the UFSAR.

Evaluation: SMARGINS is a data processing code which compares results from nuclear design calculations using SIMULATE-3 to fuel design limits, and thereby verifies acceptability of the applicable limits for the core design. SMARGINS also edits a variety of data based on SIMULATE-3 results. SMARG08 was certified in accordance with departmental guidance and verified to yield the same results as SMARG07 except for the modifications described below.

SMARG08 incorporates the following modifications:

Modifications to extrapolate SIMULATE-3 power distributions based on axial offset. SMARG07 only interpolated state points. The extrapolation is performed using the same method as interpolation. Extrapolation increases the range of power distributions used to determine the acceptability of applicable fuel design limits. Analyses using the increased axial offset space will result in a conservative evaluation of the acceptability of fuel design limits.

Modifications to assess the power peaking in the lower and upper regions of the core which are excluded from Technical Specification required monitoring in Surveillance Requirements 3.2.1.2 and 3.2.1.3.

Other miscellaneous modifications to increase the usability of the code and fix minor bugs. A complete list of modifications is provided in the SDQA report.

SAR references to this software are in DPC-NF-2010A "Nuclear Physics Methodology for Reload Design" and DPC-NE-2011PA "Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors". Catawba UFSAR Section 4.3.2.2.6 also discusses margin calculations performed by this software and references DPC-NE-201 IPA. The methodology used in some SMARGINS calculations is also discussed in DPC-NE-2009PA "Westinghouse Fuel Transition Report". The margin calculations discussed in these SAR documents are unchanged.

The SMARG08 software, and workstations where it resides, are not part of any system, structure, or component important to safety and do not directly affect any system, structure, or component. The software is not installed at the plant, but rather on workstations in the Nuclear General Office. There are no regulatory commitments associated with this software.

SMARG08 was certified in accordance with departmental guidance and verified to yield the same results as SMARG07, excepting the new modifications. The modifications were verified in accordance with departmental guidance and are in compliance with Technical Specifications and approved methods. No new methods are introduced by this code

revision. Based on the foregoing discussion, this change does not adversely affect any design bases, safety functions, safety limits, safety margins, setpoints, or core parameters. The capability to shutdown the plant and maintain it in a safe condition is unaffected. There are no unreviewed safety questions associated with this computer code revision. No Technical Specification changes are required. No UFSAR changes are required.

181 **Type:** Miscellaneous Items **Unit:** 0

Title: Technical Specification 3.8.1 "AC Sources - Operating", Basis Revision 0

Description: The Limiting Condition for Operation of Catawba Technical Specification 3.8.1, "AC Sources - Operating" states in part "The following AC Electrical Sources shall be operable:

a. Two qualified circuits between the Offsite Transmission Network and the Onsite Essential Auxiliary Power System".

The LCO basis for this section states in part "For the Offsite AC Sources, separation and independence are provided to the extent practical. A circuit may be connected to more than one ESF Bus, with fast transfer capability to the other circuit operable, and not violate separation criteria. The activity being evaluated is the deletion of the sentence " A circuit may be connected to more than one ESF bus, with fast transfer capability to the other circuit operable and not violate separation criteria., This wording originated in the Westinghouse Owner's Group Standard Technical Specifications, NUREG 1431. It was written for a plant with a different electrical design than the Catawba design. . This sentence should not have been included in the Catawba Improved Technical Specifications. The sentence will be removed to help avoid confusion and to represent the actual configuration of the plant. A similar description is not found in Chapter 8, "Electrical Power" or Chapter 15 "Accident Analysis" of the Catawba UFSAR. Therefore there are no design basis accidents that will be affected by the removal of the sentence.

Evaluation: There are no design basis accidents that will be affected by the removal of the sentence. There will be no effects on the operation or function of the offsite circuits or the ESF buses, since the design of the system will not be changed. Deletion of this sentence will make the bases more conservative by effectively removing the permissive for an offsite circuit to be connected to more than one ESF bus. There is no unreviewed safety question associated with this change. No Technical Specification changes are required. No UFSAR changes are required.

185 **Type:** Miscellaneous Items **Unit:** 0

Title: Technical Specification Surveillance Requirement 3.6.9.3 Bases Change (Hydrogen Mitigation System)

Description: The Bases of Technical Specification Surveillance Requirement 3.6.9.3 requires clarification. Specifically this clarification is that the 1700 degrees F specified in Surveillance Requirement (SR) 3.6.9.3 is required under normal plant conditions and that this temperature incorporates sufficient margin so that, under degraded bus voltage condition, the hydrogen ignitors (glow plugs) are capable of achieving the 1500 degrees F required to ignite the anticipated post-accident hydrogen concentrations. The SR Bases is to be changed as follows:

PRESENT STATEMENT:

SR 3.6.9.3

A more detailed functional test is performed every 18 months to verify system OPERABILITY. Each glow plug is visually examined to ensure that it is clean and that the electrical circuitry is energized. All ignitors (glow plugs), including normally inaccessible ignitors, are visually checked for a glow to verify that they are energized. Additionally, the surface temperature of each glow plug, is measured to be approximately 1700 degrees F to demonstrate that a temperature sufficient for ignition is achieved. The 18 month frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed at the 18 month frequency, which is based on the refueling, cycle. Therefore, the frequency was concluded to be acceptable from a reliability standpoint."

REVISED STATEMENT:

SR 3.6.9.3

A more detailed functional test is performed every 18 months to verify system OPERABILITY. Each glow plug is visually examined to ensure that it is clean and that the electrical circuitry is energized. All ignitors (glow plugs), including normally inaccessible ignitors, are visually checked for a glow to verify that they are energized. Additionally, the surface temperature of each glow plug is measured to be approximately 1700 degrees F to demonstrate that a temperature sufficient for ignition is achieved. The 1700 degree F, temperature is a surveillance requirement. "An Analysis of Hydrogen Control Measures at McGuire Nuclear Station" identifies that the required normal operation temperature is 1500 degrees F. Therefore, based upon ignitor performance testing conducted at Catawba, the surveillance requirement of 1700 degrees F ensures that sufficient margin is present for continued hydrogen ignition under degraded bus conditions. The 18 month frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed at the

18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint."

Evaluation: This clarification of the Surveillance Requirement Bases will not change the ability of the Hydrogen Mitigation System to satisfy its design basis requirements. This change will have no effect on the probability or consequences of accidents evaluated in the UFSAR.

There is no unreviewed safety question associated with this procedure revision. No Technical Specification changes are required. No UFSAR changes are required.

206 Type: Miscellaneous Items **Unit:** 0

Title: Technical Specification Surveillance requirement 3.8.1.9 Bases and Change to UFSAR Sections 8.3.1.2.4 and 8.3.1.1.3.11 to reflect revision of Regulatory Guide 1.9, PT/1(2)/A/4350/12A(B) Revision 0B, 1E, 0B, and 0C

Description: The revision of Regulatory Guide 1.9 (RG 1.9) listed in the bases of Improved Technical Specification Section 3.8.1.9 is being changed from revision 0 to revision 2. This evaluation shows that Catawba Nuclear Station is licensed to Regulatory Guide 1.9 revision 2, December 1979. Improved Technical Specification Surveillance Requirement 3.8.1.9 contains data from RG 1.9 revision 2 but the reference for the bases mistakenly lists revision 0. UFSAR Sections 8.3.1.2.4 and 8.3.1.1.3.11 also contains information from revision 0 of Regulatory Guide 1.9; therefore, this evaluation also covers the change to these sections.

The Improved Technical Specification (ITS) added additional information to the surveillance requirement for Emergency Diesel Generator load rejection. The ITS SR now includes a time for voltage and frequency to be within the acceptance range. The addition of the recovery time was consistent with NUREG-1431 Revision 1, "Standard Technical Specifications, Westinghouse Plants". Catawba Nuclear Station implemented ITS which included the bases for the recovery time. The problem occurred when the wrong revision of the Regulatory Guide was listed in the bases. The actual revision number for RG 1.9 was input by Catawba during ITS implementation based on the revision that was listed in Technical Specification Licensing Amendments 172 (Unit 1) and 164 (Unit 2). UFSAR Section 8.3.1.2.4 and 8.3.1.1.3.11 also contains information from RG 1.9 revision 0; therefore, this evaluation will also cover the change to these sections. Catawba is actually licensed to Regulatory Guide 1.9 revision 2 and the basis for the Technical Specifications Licensing Amendments 172 (Unit 1) and 164 (Unit 2) and the UFSAR were not properly revised when revision 2 was adopted in December 1979.

Evaluation: There are no unreviewed safety questions associated with this UFSAR Revision. This revision has no effect on the probability or consequences of accidents evaluated in the UFSAR. No Technical Specification changes are required. A change is required for UFSAR Sections 8.3.1.2.4 and 8.3.1.1.3.11.

123 **Type:** Miscellaneous Items **Unit:** 0

Title: Temporary Modification CNTM-0043, Tie-in to Demineralized Water System sample line for chemical addition

Description: Temporary Modification CNTM-0043 adds a tee and an isolation valve to the Demineralized Water System at the outlet of valve 1YM-194 (Makeup Vacuum Deaerator A Discharge Sample). This modification will allow Chemistry to add wet lay-up chemicals to the discharge of the deaerator which will be in discharge to the Steam Generators.

The vacuum deaerators for the Demineralized Water System are designed to supply deaerated water to the Steam Generators and to the Reactor Makeup Water Storage Tank. Normally, the Steam Generators are filled during refueling outages through the Auxiliary Feedwater System with chemical additions made by the Chemical Addition system through the Steam Generator Wet Lay-up system. During the 1EOC12 refueling outage, there are anticipated times when these systems will not be available. In order to minimize personnel dose and maintain a corrosion preventive environment in the generators, this modification will be performed.

Evaluation: The addition of water via the Demineralized Water system has been previously analyzed and is a design function of the system. The addition of chemicals in this manner has not been addressed. The chemicals that will be added are the same chemicals that are normally fed through the Chemical Addition system, carbohydrazide and 3-methoxypropylamine (3-MPA). Injecting these chemicals in the Water Treatment Room will allow thorough mixing prior to reaching the Steam Generators. The 10CFR50.59 evaluation for the required Chemistry operating procedure for use of this temporary mod will evaluate the use of this flow path, including system alignments and recovery.

Per analysis by the Technical Services Chemistry section, should the entire volume required for addition to any Steam Generator be transferred to the Reactor Makeup Water Storage Tank, there would be no effect on the plant. Per a previous evaluation, "If the Reactor Makeup Water Storage Tank were less than full, the resulting concentrations and conductivity would be even higher. Since the lay-up chemicals are not specifically harmful to reactor coolant system materials, meeting the conductivity specification will be sufficient removal".

There are no unreviewed safety questions associated with this activity. This change will have no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required.

20 **Type:** Miscellaneous Items **Unit:** 2

Title: Temporary Modification CNTM-0020, "Temporary Isolation of the 1B Stage of Heater 2HETR0528"

Description: During the 10-hour heater surveillance per procedure PT/2/A/4450/001 for the Unit 2 Containment Purge Ventilation System filter units (CPFU-2A and CPFU-2B), a problem was discovered with the inlet duct heater (2HETRO528). The heater was not producing the normal temperature rise due to a fault in the first stage 1B element. Repairing the heater is difficult because no replacement elements are currently availlable and there is a long lead-time for procurement of new elements. Temporary Modification CNTM-0020 was generated to isolate the bad element and allow the other elements in the heater to operate because of the need to run the heater when the filter units are operating.

Operation of the inlet duct heater without the 1B stage will reduce the heater capacity from 120 kW to 100 kW. (The first stage 1A element is 20 kW, 1B is 20 kW and stages 2, 3 and 4 are 26.6 kW each.) Operation in this degraded condition does not affect the operability of the Containment Purge Ventilation System because the heater is not a required system component. The heater was originally installed to ensure that the relative humidity of the air entering the filter units was less than or equal to 70%. However, with the adoption of the carbon testing methodology of ASTM D3803-89, control of relative humidity was no longer necessary. Use of the ASTM D-3803-89 methodology was reviewed and accepted by the NRC and is now included in the Catawba Technical Specifications in Section 5.5.11. The acknowledgement that the heater is not required for operability is included in the Technical Specifications under Required Action B.2 of Section 3.9.3. Since moisture tends to degrade the life of carbon, the heater has been retained in the system only to extend the life of the carbon.

Evaluation: Since the heater is not required to ensure filter unit operability, operation without any heater capacity or operation with only partial heater capacity (100 kW out of 120 kW) is not a concern. This is true whether one or both of the filters are in service. The heater controls will still operate as designed (except that the 1B element will not be energized) to aid in preserving carbon media life. A work order was generated to investigate the degraded heater. There are no unreviewed safety questions identified with revision 1 to PT/0/A/4450/020, Ventilation Filter Testing Program (VFTP). No Technical Specification changes are required. No UFSAR changes are required .

14 Type: Miscellaneous Items **Unit:** 1

Title: Temporary Modification CNTM-0022, "Operating the 1A Emergency Diesel Generator with the channel cards for the Right and Left Bank Low Turbocharger Oil pressure trip removed from the Rosemount Non-Emergency Trip Panel.

Description: The channel cards for the Right and Left Bank Low Turbocharger Oil Pressure Trip have failed in the Rosemount Non-emergency trip panel on the 2B Emergency Diesel Generator. At the time of the failure no replacement cards were immediately available. Low turbocharger oil pressure is a non-emergency trip, which is bypassed during an emergency run.

A decision was made to remove the respective cards from this Diesel Generator and place them in the trip panel on the 2B Diesel Generator. The reason the cards are being moved to the 2B Diesel Generator is to ensure all protective trips are available for the post maintenance break-in runs following 2EOC10 inspections.

Evaluation: The emergency diesel generators used to provide emergency power at Catawba Nuclear Station are designed with turbochargers to support the fast starting and loading requirements. The turbochargers require a constant supply of lube oil for bearing lubrication. The turbochargers are required to support emergency diesel generator operation and are thus classified as nuclear safety related components. The oil supply is also part of a nuclear safety related system. The turbochargers are designed with a low oil pressure alarm (20 psi) and a low oil pressure trip (15 psi). The alarm and trip are provided for reliability concerns, not for operability.

An "Operable But Degraded/Non Conforming Evaluation" was performed to show that the 1A Emergency Diesel Generator is operable without the Low Lube Oil turbocharger pressure trips.

The purpose of the trip panel is to receive input from pressure transmitters and then send a signal to the respective alarm and to trip the engine. The only signals to go through the panel are the non-emergency trips which are bypassed on an emergency start signal. Removing the two cards which receive signals from pressure transmitters 1LDPT5220 and 1LDPT5230, will not affect the function of the other channels. The removed cards create an open circuit which will not affect operation of the other functions nor will it affect the portion of the circuit which bypasses the trip functions on an Emergency Start Signal.

Operating the 1A Emergency Diesel Generator with the low lube oil turbocharger pressure trip cards removed from the Rosemount non-emergency trip panel does not create an unreviewed safety question. The 1A DG will continue to provide Emergency AC Power in response to a Design Basis Event as required per Technical Specifications. The non-emergency trips will continue to be bypassed as required on an emergency start signal.

The Diesel Generators are not accident initiators as evaluated in the UFSAR. Therefore, removing the cards does not increase the probability or consequences of accidents evaluated in the UFSAR.

No Technical Specification changes are required. No UFSAR changes are required

134 **Type:** Miscellaneous Items **Unit:** 2

Title: Temporary Modification CNTM-0035 to provide an alternate cooling water supply source to the Unit 2A Diesel Generator Jacket Water Heat Exchanger

Description: The purpose of this Temporary Modification is to provide an alternate cooling water supply source to the Unit 2A Diesel Generator Jacket Water Heat Exchanger during Nuclear Service Water System A Train supply pipe cleaning work during the 1EOC12 Refueling Outage. The alternate flowpath is from Fire Hydrant 0RYH031 to the heat exchanger via 2 1/2 inch fire hose and discharging through the normal Nuclear Service Water System/Low Pressure Service Water System flowpath. A temporary flange will be installed on the inlet of the Diesel Generator 2A Engine Cooling Water System heat exchanger when supply piping is disconnected for pipe cleaning. The fire hose will be routed from the hydrant in the yard through a 4 inch gedney plug on the diesel building roof to this flange. The Temporary Modification will be installed during the Nuclear Service Water System A Train supply pipe cleaning when the normal Diesel Generator cooling water supply is unavailable.

Evaluation: This temporary modification will install 2 1/2 inch fire hose between the hydrant 0RYH031 and the Unit 2A Diesel Generator Jacket Water Heat Exchanger. This will establish cooling flow through the Unit 2A Diesel Generator Engine Cooling Water heat exchanger thereby allowing the 2A Diesel Generator to be maintained available for manual starting and loading if desired by Operations. An Engineering calculation shows that the supply flow to the heat exchanger is sufficient for full load engine operation. Sufficient flow can be obtained from the flow path as described above without affecting the design basis of the Interior/Exterior Fire Protection System.

Fire hydrant 0RYFH031 will be used as the source for the alternate cooling. The fire hose will be connected to valves 1RY-113 and 1RY-114. Gated Y connections or equivalents will be installed on the outlet valves of 0RYFH31 to allow the hydrant to remain in service with the temporary modification installed. The temporary modification will be implemented and ready to be placed in service per procedure if diesel generator operation is desired by Operations. Procedural guidance of the temporary modification ensures the fire hose will only be charged when Operations is running the diesel generator thereby allowing the engine and the water source to be secured if a leak in the hose occurs. This will eliminate flooding concerns. Furthermore, flooding in the diesel room is not a concern because the respective diesel will be in an LCO during the Nuclear Service Water System pipe cleaning as allowed per Technical Specification and not relied upon to supply emergency AC power.

The fire hose will be routed to the Diesel Generator heat exchanger via 4 inch gedney plugs in the diesel building roofs. The Diesel Generator intakes and walls around a nearby staircase as well as other structures protect to a large extent the roof area where the pipe sleeve penetrations are located from the horizontal component of potential "tornado missiles."

The pipe elbows that are provided (as described above) give the inside of the diesel generator rooms approximately equivalent tornado missile protection as is provided when the normal configuration is in place on the roof of the building.

Based on the information discussed above, the alternate cooling water flow path will allow the 2A diesel generator to operate at full load if desired without affecting the operability of the Fire Protection Systems.

The 2A diesel generator will be technically inoperable during the period of time this modification is installed. Unit 2 A Train Emergency AC Power will be in an LCO as allowed per Technical Specifications during the Nuclear Service Water System pipe cleaning and thus will not be relied on to respond to design bases events. The Temporary Modification is being implemented to maintain the technically inoperable 2A DG available.

The diesel generators are not accident initiators as evaluated in the UFSAR. Therefore, providing an alternate cooling water supply for it does not increase the probability of accidents evaluated in the UFSAR. There are no unreviewed safety questions associated with this temporary modification. No Technical Specification changes are required. No UFSAR changes are required.

135 **Type:** Miscellaneous Items **Unit:** 2

Title: Temporary Modification CNTM-0036 to provide an alternate cooling water supply source to the Unit 2A Diesel Generator Jacket Water Heat Exchanger

Description: The purpose of this Temporary Modification is to provide an alternate cooling water supply source to the Unit 2B Diesel Generator Jacket Water Heat Exchanger during Nuclear Service Water System A Train supply pipe cleaning work during the 1EOC12 Refueling Outage. The alternate flowpath is from Fire Hydrant 0RYH031 to the heat exchanger via 2 1/2 inch fire hose and discharging through the normal Nuclear Service Water System/Low Pressure Service Water System flowpath. A temporary flange will be installed on the inlet of the Diesel Generator 2B Engine Cooling Water System heat exchanger when supply piping is disconnected for pipe cleaning. The fire hose will be routed from the hydrant in the yard through a 4 inch gedney plug on the diesel building roof to this flange. The Temporary Modification will be installed during the RN A Train supply pipe cleaning when the normal Diesel Generator cooling water supply is unavailable.

Evaluation: This temporary modification will install 2 1/2 inch fire hose between the hydrant 0RYH031 and the Unit 2B Diesel Generator Jacket Water Heat Exchanger. This will establish cooling flow through the Unit 2B Diesel Generator Engine Cooling Water heat exchanger thereby allowing the 2B Diesel Generator to be maintained available for manual starting and loading if desired by Operations. An Engineering calculation shows that the supply flow to the heat exchanger is sufficient for full load engine operation. Sufficient flow can be obtained from the flow path as described above without affecting the design basis of the Interior/Exterior Fire Protection System.

Fire hydrant 0RYFH031 will be used as the source for the alternate cooling. The fire hose will be connected to valves 1RY-113 and 1RY-114. Gated Y connections or equivalents will be installed on the outlet valves of 0RYFH31 to allow the hydrant to remain in service with the temporary modification installed. The temporary modification will be implemented and ready to be placed in service per procedure if diesel generator operation is desired by Operations. Procedural guidance of the temporary modification ensures the fire hose will only be charged when Operations is running the diesel generator thereby allowing the engine and the water source to be secured if a leak in the hose occurs. This will eliminate flooding concerns. Furthermore, flooding in the diesel room is not a concern because the respective diesel will be in an LCO during the Nuclear Service Water System pipe cleaning as allowed per Technical Specification and not relied upon to supply emergency AC power.

The fire hose will be routed to the Diesel Generator heat exchanger via 4 inch gedney plugs in the diesel building roofs. The Diesel Generator intakes and walls around a nearby staircase as well as other structures protect to a large extent the roof area where the pipe sleeve penetrations are located from the horizontal component of potential "tornado missiles."

The pipe elbows that are provided (as described above) give the inside of the diesel generator rooms approximately equivalent tornado missile protection as is provided when the normal configuration is in place on the roof of the building.

Based on the information discussed above, the alternate cooling water flow path will allow the 2B diesel generator to operate at full load if desired without affecting the operability of the Fire Protection Systems.

The 2B diesel generator will be technically inoperable during the period of time this modification is installed. Unit 2 B Train Emergency AC Power will be in an LCO as allowed per Technical Specifications during the Nuclear Service Water System pipe cleaning and thus will not be relied on to respond to design bases events. The TM is being implemented to maintain the technically inoperable 2B DG available.

The diesel generators are not accident initiators as evaluated in the UFSAR. Therefore, providing an alternate cooling water supply for it does not increase the probability of accidents evaluated in the UFSAR. There are no unreviewed safety questions associated with this temporary modification. No Technical Specification changes are required. No UFSAR changes are required.

147 **Type:** Miscellaneous Items **Unit:** 2

Title: Temporary Modification CNTM-0039 to provide an alternate cooling water supply source to the Unit 2A Diesel Generator 2A1 Starting Air Aftercooler

Description: Temporary Modification CNTM-0039 will provide an alternate cooling water supply source to the Unit 2A Diesel Generator 2A1 Starting Air Aftercooler during the time that the A Train Nuclear Service Water System is undergoing pipe cleaning work during the 1EOC12 Refueling Outage. The alternate flowpath is from the Diesel Generator 2A eyewash and shower station to the cooler drain valve 2RN-946 via temporary hose, through the aftercooler, and to the Diesel Generator Room Sump via the aftercooler vent valve 2RN-947 discharging to a floor drain.

Installation of the Temporary Modification will consist of connecting hose from the shower station to the VG 2A1 Drain Valve 2RN-946 and connecting tubing from the VG 2A1 aftercooler vent valve 2RN-947 to the Diesel Room Sump Pump System Sump. This temporary modification will only be installed during the cleaning of the A Train Nuclear Service Water System piping (when the normal aftercooler supply source is unavailable).

This Temporary Modification will be required to maintain the Diesel Generator 2A Starting Air Tanks charged during the Nuclear Service Water A Train supply pipe cleaning during the 1EOC12 Refueling Outage. During A Train supply pipe cleaning, the 2A Diesel Generator will be maintained "available" per Temporary Modification CNTM-0035 which will require that the Diesel Generator Starting Air System Storage Tanks to remain charged.

Evaluation: This temporary modification will install hose from the Diesel Generator 2A emergency shower station to the Diesel Generator Engine Starting Air System 2A1 aftercooler drain valve 2RN-946 and tygon tubing from the aftercooler vent valve 2RN-947 to the Diesel Room Sump System sump via a floor drain. This work will establish cooling flow through the 2A1 aftercooler thereby allowing the nonsafety related Diesel Generator Engine Starting Air System 2A1 compressor and associated components to operate. Per calculation CNC-1223.5907-0005, the supply flow at the shower station is 25 gpm. There will be some piping losses through the hose, cooler, and fittings, although, per engineering judgment the flow to the aftercooler will be greater than the preferred minimum of 6.5 gpm when including the assumed flow loss. Therefore the 2A1 aftercooler can maintain the required output air temperature without affecting the operation of the desiccant drying tower and subsequent starting air quality. Sufficient flow can be obtained from the flow path as described above without affecting the non-safety related function of the 2A1 Diesel Generator Starting Air Compressor, Cooler, and Drying Tower.

The connection at the shower station will be installed so that the shower station remains available. Therefore, installation of the temporary hose will not affect any personnel safety requirements because the shower station will remain available.

The Diesel Generator Room Sump Pumps are nuclear safety related components with a capacity of 50 gpm each. If input to the sump cannot be maintained by one pump the second pump will start increasing the capacity to 100 gpm. Therefore, the additional input (less than 25 gpm) to the sump will not increase the potential of Diesel Generator room

flooding because this input is less than the capacity of the sump pumps. If one pump fails, the second pump can maintain the sump input from normal system leakage and the less than 25 gpm cooler flow discharge. Furthermore, when one Diesel Generator Room Sump Pump is out of service Operations performs increased surveillance to ensure room flooding does not occur. Therefore, no system, structure, or components addressed in the UFSAR will be affected in any significant manner due to potential flooding.

This temporary modification does not replace any components but only establishes an alternate 2A1 Diesel Generator Engine Starting Air System aftercooler flowpath to allow continued operation of the non-safety related Diesel Generator 2A1 Starting Air compressor and associated components while the normal cooling water flow is not available. The purpose of the temporary modification is to allow operation of the Diesel Generator 2A1 Starting Air aftercooler while the normal cooling water supply source is unavailable.

There are no unreviewed safety questions associated with this temporary modification. All of this equipment will be declared inoperable during this evolution. This temporary modification will only to be in place to make the associated diesel generator "available". No Technical Specification changes are required. No UFSAR changes are required.

148 **Type:** Miscellaneous Items **Unit:** 2

Title: Temporary Modification CNTM-0040 to provide an alternate cooling water supply source to the Unit 2B Diesel Generator 2B1 Starting Air Aftercooler

Description: Temporary Modification CNTM-0040 will provide an alternate cooling water supply source to the Unit 2B Diesel Generator 2B1 Starting Air Aftercooler during the time that the B Train Nuclear Service Water System is undergoing pipe cleaning work during the 1EOC12 Refueling Outage. The alternate flowpath is from the Diesel Generator 2B eyewash and shower station to the cooler drain valve 2RN-957 via temporary hose, through the aftercooler, and to the Diesel Generator Room Sump via the aftercooler vent valve 2RN-958 discharging to a floor drain.

Installation of the Temporary Modification will consist of connecting hose from the shower station to the VG 2B1 Drain Valve 2RN-957 and connecting tubing from the VG 2B1 aftercooler Vent Valve 2RN-958 to the Diesel Room Sump Pump System sump. This temporary modification will only be installed during the cleaning of the B Train Nuclear Service Water System piping (when the normal aftercooler supply source is unavailable).

This Temporary Modification will be required to maintain the Diesel Generator 2B Starting Air Tanks charged during the Nuclear Service Water B Train supply pipe cleaning during the 1EOC12 Refueling Outage. During B Train supply pipe cleaning, the 2B Diesel Generator will be maintained "available" per Temporary Modification CNTM-0036 which will require that the Diesel Generator Starting Air System Storage Tanks to remain charged.

Evaluation: This temporary modification will install hose from the Diesel Generator 2B emergency shower station to the Diesel Generator Engine Starting Air System 2B1 aftercooler drain valve 2RN-957 and tygon tubing from the aftercooler vent valve 2RN-958 to the Diesel Room Sump System sump via a floor drain. This work will establish cooling flow through the 2B1 aftercooler thereby allowing the nonsafety related Diesel Generator Engine Starting Air System 2B1 compressor and associated components to operate. Per calculation CNC-1223.5907-0005, the supply flow at the shower station is 25 gpm. There will be some piping losses through the hose, cooler, and fittings, although, per engineering judgment the flow to the aftercooler will be greater than the preferred minimum of 6.5 gpm when including the assumed flow loss. Therefore the 2B1 aftercooler can maintain the required output air temperature without affecting the operation of the desiccant drying tower and subsequent starting air quality. Sufficient flow can be obtained from the flow path as described above without affecting the non-safety related function of the 2B1 Diesel Generator Starting Air Compressor, Cooler, and Drying Tower.

The connection at the shower station will be installed so that the shower station remains available. Therefore, installation of the temporary hose will not affect any personnel safety requirements because the shower station will remain available.

The Diesel Generator Room Sump Pumps are nuclear safety related components with a capacity of 50 gpm each. If input to the sump cannot be maintained by one pump the second pump will start increasing the capacity to 100 gpm. Therefore, the additional input

(less than 25 gpm) to the sump will not increase the potential of Diesel Generator room flooding because this input is less than the capacity of the sump pumps. If one pump fails, the second pump can maintain the sump input from normal system leakage and the less than 25 gpm cooler flow discharge. Furthermore, when one Diesel Generator Room Sump Pump is out of service Operations performs increased surveillance to ensure room flooding does not occur. Therefore, no system, structure, or components addressed in the UFSAR will be affected in any significant manner due to potential flooding.

This temporary modification does not replace any components but only establishes an alternate 2B1 Diesel Generator Engine Starting Air System aftercooler flowpath to allow continued operation of the non-safety related Diesel Generator 2B1 Starting Air compressor and associated components while the normal cooling water flow is not available. The purpose of the temporary modification is to allow operation of the Diesel Generator 2B1 Starting Air aftercooler while the normal cooling water supply source is unavailable.

There are no unreviewed safety questions associated with this temporary modification. All of this equipment will be declared inoperable during this evolution. This temporary modification will only to be in place to make the associated diesel generator "available". No Technical Specification changes are required. No UFSAR changes are required.

95 **Type:** Miscellaneous Items **Unit:** 1

Title: Temporary Modification Work Order 98263152

Description: This temporary modification will delete the input from the Reactor Coolant Pump 1C lower bearing water temperature to the Control Room Annunciator 1AD-7B/2. Annunciator 1 AD-7 B/2 provides indication that any of the Unit 1 Reactor Coolant Pump lower bearing water high temperature alarms has occurred (170 degrees F). There is a reference to procedure AP/l/A/5500/08, Malfunction of Reactor Coolant Pump, in the response to this annunciator. Analog meters provide Control Room Operators continuous indication of the Reactor Coolant Pump lower bearing water temperatures. This indication is also incorrect and instrument 1NVP5350 is marked out of service. Indication is also provided for the # 1 seal leak-off temperature. In the Control Room Operator's assessment of the situation involving a lower bearing high temperature, alarm consideration is given also to the # 1 seal leak-off temperature, the seal injection flow rate, pump / motor vibration, and thermal barrier cooling via Component Cooling System flow and temperatures.

The measurements feeding the Reactor Coolant Pump lower bearing temperatures are provided by Westinghouse supplied RTDs measuring water temperature just above the pump bearings. This water is provided as seal injection water just below the pump bearings but above the thermal barrier heat exchanger. Depending on the seal leak-off flow, approximately three gallons per minute are flowing from below the bearing up through and around the bearing and into the # 1 pump seal. In accordance with the design of the controlled leak-off seal, the great majority of the seal water exits the pump via the # 1 seal leak-off back to the Volume Control Tank. The temperature of the #1 seal leak-off is measured with a second RTD positioned above the #1 seal, inside the lower seal housing and below the upper seal housing (below the # 2 seal). Annunciator 1AD-7 B/I provides indication that any of the Unit 1 Reactor Coolant Pump #1 seal outlet temp alarms has occurred (200 degrees F). The Operator Aid Computer (OAC) Hi Alarm for the #1 seal leak-off is set at 190 degrees F. with the Hi-Hi alarm set at 235 degrees F.

This temporary modification will defeat the input to the Reactor Coolant Pump lower bearing water temperature high alarm annunciator for the 1C Reactor Coolant Pump. Current plans are to replace the RTD in Reactor Coolant Pump 1C during the next Unit 1 refueling outage, 1EOC12. This Temporary Modification will be removed at that time.

Reactor Coolant Pump 1C pump lower bearing water temperature indication began increasing in temperature, but other parameters that should show changes associated with this observation did not change. This suggests that the indication is in error. Increased RTD resistance would result in the indication noted.

Operator Aid Computer point C1A0837, Reactor Coolant Pump 1C Lower Bearing Water Temperature, began gradually climbing from below 140 degrees F in the evening of 4/l/00. In the evening of 4/2/00 the increase accelerated and took a step change up to above 150 degrees F. The increasing trend continued at the original pace until another step change took place in the evening of 4/3/00 which took the temperature indication above 170 degrees F (the alarm value). The shut down value for this parameter is 225 degrees F.

All other parameters that are physically linked to the lower bearing water temperature , directly and indirectly, are not changing.

Evaluation: This temporary modification will defeat the Reactor Coolant Pump 1C input to annunciator 1AD-7 B/2. The control room operators have continuous indication of the # 1 seal outlet water temperature via meters. The annunciator 1AD-7 B/I is not being altered and is based on the temperature of the same water made further down stream. It will remain in service providing unaltered alarm of the same water at temperatures well below the maximum allowed temperatures. There are no unreviewed safety questions associated with this temporary modification. No Technical Specification changes are required. No UFSAR changes are required.

58 Type: Miscellaneous Items **Unit:** 2

Title: Temporary Station Modification (TSM) Work Order 98204831 (01/02) and Procedure Changes for OP/2/A/6100/002 Rev 120: Bypass for P-12 Interlock in Mode 4 for Extended Cooldown on Condenser Steam Dumps.

Description: Temporary Station Modification (TSM) Work Order 98204831 provides the method for bypass of the P-12 interlock and provides a method to use additional condenser steam dump valves for unit cooldown while in Operations Procedure OP/2/A/6100/002. The "P-12" interlock will be bypassed in the Auxiliary Safeguards Cabinets to disable the interlock when appropriate pressure and temperature conditions are met during Unit 2 cooldown in Mode 4. Technical Specification 3.3.2 requires that the interlock be operable during Modes 1, 2, and 3. This interlock may be bypassed when the unit is in Mode 4 since it is no longer required by Technical Specifications. The condenser steam dumps are controlled using the Steam Pressure Controller before and after the P-12 interlock is bypassed. This controller can be operated in auto with a steam pressure setpoint or in manual with a pushbutton demand signal. This procedure reduces the amount of time the Residual Heat Removal system is needed to operate during unit cooldown by performing an extended cooldown using condenser dump valves at lower temperatures. This method of cooldown is expected to reduce the amount of crud precipitated upon start of the Residual Heat Removal System and lower general area dose rates during shutdown.

There are two major issues to be considered for this change.

1. The ability to add positive reactivity at a faster rate than would be possible using only three Main Steam Bypass to Condenser System valves, will be provided by the additional cooldown capacity afforded by the six additional dump valves' heat removal capability at the rector coolant system temperature at which the TSM is installed (300 deg F. or below).

   Procedure OP/2/A/6100/002 Rev 120 has been revised to include provisions for either Mode 6 boron concentration or the boron concentration associated with the final temperature of the cooldown prior to installing the TSM to bypass the P-12 interlock. Thus, adequate shutdown margin will be maintained and return to criticality is not possible.

2. The ability to cool the Reactor Coolant System and potentially challenge the Technical Specification cooldown limit curve for Unit 2 given in Technical Specification Figure 3.4.3-2 will be afforded by the six additional dump valves' heat removal capability.

An evaluation was performed to assess the cooldown potential following failure of the steam dump controller after placing the TSM in service (Reactor Coolant System Tavg 300 deg F or below) . It was determined that the Tech Spec cooldown limit of 100 deg F/hour should not be violated due to this failure alone with all nine Main Steam Bypass to Condenser System valves open. It was also shown that an existing failure mode of the Residual Heat Removal flow control valve failing open would lead to a cooldown rate

more severe than a failure of the steam dump controller at these temperatures/pressures. Failure of the steam dump controller at Reactor Coolant System temperatures just below the P-12 setpoint (553 degrees F.) and the associated opening of just one bank (Bank #1) of three Main Steam Bypass to Condenser System valves results in a much worse cooldown by comparison. Thus, PTS events will not be exacerbated by this alternate cooldown method.

Evaluation: Overcooling events involving increased heat removal by the secondary system are analyzed in section 15.1 of the UFSAR. These include Main Feedwater System malfunctions associated with a decrease in Main Feedwater System temperature (15.1.1), increase in Main Feedwater System flow (15.1.2), increase in steam flow (15.1.3), inadvertent opening of a Steam Generator Power Operated Relief Valve (PORV) or safety valve, and Steam Line Break (15.1.5). The procedure changes discussed and the Temporary Station Modification installed do not affect the analysis of any of these events and do not increase the probability of any event analyzed in the UFSAR. These changes only affect the means by which a cooldown is accomplished below Mode 3. None of these changes are accident initiators. Consideration has been given to possible malfunction of the steam dump controller to fail to maximum output and open all steam dump valves. It is not anticipated that the Technical Specification cooldown limit of 100 degrees F/hour would be achieved.

The limiting Steam Line Break is analyzed at Reactor Coolant System conditions of no-load temperature and Mode 3 reactivity (- 1300 pcm). As noted in the bases of the Technical Specifications the protection provided by the P-12 interlock in not needed in Mode 4 (or below) because "there is insufficient energy contained in the secondary side of the Unit to have an accident". All UFSAR analyses remain bounding considering this change.

Since the reactor is shutdown, and the boron concentration associated with the most limiting temperature of the cooldown prior to installing the Temporary Station Modification is established prior to invoking this alternate cooldown method using steam dump valves, the possibility of an overcooling event involving return to criticality is not credible at this phase of shutdown operation. The Residual Heat Removal System is available to prevent uncontrolled heatup of the Reactor Coolant System. The Steam Generator Power Operated Relief Valves (PORVs) are also available as a means of primary cooling via the secondary Main Steam System. Thus, the probability of an uncontrolled heat up due to failure of a cooling system or component (e.g. steam dumps) is not increased. No other reactivity related accident such as "Rod Ejection" (UFSAR 15.4.8), RCCA Misoperation (UFSAR 15.4.3), and "Uncontrolled RCCA Withdrawal" (UFSAR 15.4.1, 15.4.2) are affected by this Temporary Station Modification. Therefore, the probability of an accident evaluated in the UFSAR is not increased.

There are no unreviewed safety questions associated with the use of Temporary Station Modification Work Order 98204831 and associated procedure OP/2/A/6100/002 Rev 120. No Technical Specification changes are required. No UFSAR changes are required.

87 **Type:** Miscellaneous Items **Unit:** 2

Title: Temporary Station Modification CNTM-0028, Install gag on Valve 2RC-31

Description: Work Order 98283718 will repair a problem on the operator for valve 2RC-31, Cooling Tower 2A make-up water flow control valve. While the operator is being repaired, the make-up flow path through 2RC-31 will need to be in service. This Temporary Station Modification will install a gag on 2RC-31 in its fully open position to allow the Cooling Tower 2A to remain in service. Chemistry personnel will control flow using the manual butterfly valve 2RC-95, upstream of valve 2RC-31, and Procedure OP/0/B/6400/017.

The gag was fabricated from an old operator drive bushing, steel plate, and pipe. The plate bolts up to the existing operator mounting holes in the valve body. The pipe is welded to the center of the plate and fits over the valve stem. The drive bushing is welded to the other end of the pipe in a position such that a key can be placed in the valve shaft keyway and mate with the bushing keyway. This arrangement will be sufficient to maintain the valve in the full open position while in service. The valve and operator assembly weigh considerably more than the drive bushing gag, so there is no concern with the weight of the gag. No seismic or civil concerns were identified.

While the gag is installed under this Temporary Station Modification, make-up to Cooling Tower 2B will be in normal service with automatic make-up flow control provided by valve 2RC-32.

Evaluation: Valve 2RC-31 is not required to close during an accident, nor is it an initiator of any accident. Although the automatic level control function provided by 2RC-31 will not be available, the level will be manually controlled by valve 2RC-95. Cooling tower level control has no effect on any accident evaluated in the UFSAR. Therefore, gagging the valve open will not increase the probability of an accident.

There are no unreviewed safety questions associated with this Temporary Station Modification. No Technical Specification changes are required. No UFSAR changes are required.

154 **Type:** Miscellaneous Items **Unit:** 0

Title: Temporary Station Modification CNTM-0052 and Procedure TN/1/A/0052/TM/001, Implementation Procedure for Temporary Modification CNTM-0052, Revision 0

Description: Valve 1RN-003A (Nuclear Service Water Pump Pit A Intake from the Standby Nuclear Service Water Pond Isolation Valve) is currently stuck in the open position. This requires the A Train of the Nuclear Service Water System to be aligned to the Standby Nuclear Service Water Pond to maintain system operability. This Temporary Station Modification and Implementing Procedure will remove the valve from the system for repairs. The valve must be closed to approximately 40 degrees open to clear the hatch plug during removal.

The A Train Nuclear Service Water System Pumps are currently aligned to the Standby Nuclear Service Water Pond. The white tag for valve 1RN-003A will be removed and the actuator and extension shaft will be disconnected from the valve. A section of supported discharge piping will be removed from valve 1RN-003A to facilitate the valve's removal. This eight foot stub will be left in the Nuclear Service Water System Pump Pit. Valve 1RN-003A will be partially stroked closed and then removed from the system per Temporary Station Modification CNTM-0052 and Procedure TN/1/A/0052/TM/001, "Implementation Procedure for Temporary Modification CNTM-0052". (Stroking the valve partially closed will provide the necessary clearance to remove the valve through the hatch opening.) The Train A Nuclear Service Water System Pumps will not be running during this evolution, but will be capable of starting. The Nuclear Service Water System Pumps will maintain an available suction source from the Standby Nuclear Service Water Pond throughout this evolution.

The Nuclear Service Water System Design Basis Document states that with valve 1RN-003A open and incapable of closing, the operability of the system's A loop is maintained, but normal operating procedures are affected. With valve 1RN-003A open and incapable of closing, the A loop of the Nuclear Service System (suction and discharge) should be aligned to the Standby Nuclear Service Water Pond to prevent loss of Pond inventory to Lake Wylie. Also, operability of the Standby Nuclear Service Water Pond may be affected under Technical Specification 3.7.9.

Evaluation: Effect on Nuclear Service Water System Flow

This temporary modification and procedure alters the Nuclear Service Water System by allowing partial closing and subsequent removal of valve 1RN-003A, the train A Nuclear Service Water System Pumps inlet isolation valve from the Standby Nuclear Service Water Pond. The Operability of the Nuclear Service Water System Train A is assured as long as the valve is maintained greater than 40 degrees open or is removed from the system.

With the valve at 40 degrees open with an assumed Nuclear Service Water System flow from the A Pit of 25,000 gpm, the piping resistance changes will cause the Nuclear Service System A Pit level to decrease by approximately 0.6 feet, in addition to the 1.5 feet previously described in the 10CFR50.59 evaluation for maintaining Nuclear Service Water System A Train Operability. This additional 0.6 foot change in pit level is within all valve and pit level design parameters.

The change in piping resistance caused by closing the valve 50 degrees from the full open position would cause level in Nuclear Service Water System Pit A to decrease by an additional 0.6 feet with a assumed flow rate of 25,000 gpm. Lower flow rates would cause a smaller change in pit level. Similarly, complete removal of valve 1RN-003A reduces the overall suction piping resistance from the Standby Nuclear Service Water Pond.

Additional Areas of Review for Effect on the Nuclear Service Water System.

The process of jacking valve 1RN-003A in the closed direction may damage the valve. This damage will be evaluated and repaired prior to placing the valve back into the system and therefore need not be reviewed in this evaluation. If at any time, the disc for 1RN-003A is closed more than 50 degrees from the open direction then the Nuclear Service Water System A Train will be declared inoperable.

The additional loads placed on the support piping while the extension shaft is being removed or while valve 1RN-003A is being jacked closed and removed are within all design parameters for the Nuclear Service Water System piping from the Standby Nuclear Service Water Pond to the Nuclear Service Water System Pump Pit. Also, there are no seismic or buoyancy concerns associated with the remaining Nuclear Service Water System piping and the eight foot stub of discharge piping after the removal of the valve.

The Nuclear Service Water System is designed to supply the cooling water requirements of a simultaneous LOCA on one unit and cooldown on the other unit assuming a single failure anywhere on the system, loss of offsite power, and loss of Lake Wylie.

This temporary modification does not affect the ability of the Nuclear Service Water System to supply cooling water from the Standby Nuclear Service Water Pond. This temporary modification does not affect the likelihood of a LOCA, loss of Lake Wylie, or a loss of offsite power.

There are no alterations to the material or construction standards due to implementation of this temporary procedure.

No instrumentation on the Nuclear Service Water System is being replaced.

Operations procedures provide guidance to ensure that the Nuclear Service Water System is not operated at higher than normal pressures and minimize the time that the pumps are operated in low flow conditions.

The Nuclear Service Water System will see no active change if the Nuclear Service Water System pumps do not auto start. If an auto start occurs the only system change will be an additional 0.6 foot drop in the Train A Nuclear Service Water Pump pit level, due to the slight additional flow induced delta pressure across valve 1RN-003A. The 0.6 feet is in addition to the 1.5 feet in flow losses when two Nuclear Service Water pumps are in service. The Train A pit swap set point is 571.6 feet, and the normal Standby Nuclear Service Water Pond elevation is 574.0 feet. A pit swap is not likely when one A Train Nuclear Service Water Pump is started, but will likely occur when both pumps in the pit start in response to Ss, LOOP, or low-low (emergency low) pit level. This is within the design basis of the Nuclear Service Water System. Nuclear Service Water Pump net

positive suction head limits (elevation 555.4 feet) are not approached. Besides a LOCA with concurrent LOOP, the Nuclear Service Water System design basis includes a loss of Lake Wylie with subsequent worst-case single failure on the Nuclear Service Water System, the failure of one train of Nuclear Service Water.

Limits for the maximum vertical lift (displacement) and the maximum vertical loading on the extension shaft are incorporated into the procedure so as not to challenge any valve extension shaft limits.

No Nuclear Service Water System interface changes occur while performing this temporary procedure.

Valve 1RN-003A and the discharge stub piping are the only equipment being manipulated during execution of this procedure. There are checks to ensure that the valve will not be closed beyond the point analyzed by this 10CFR50.59 evaluation. The procedure requires verification of valve position by positive means to ensure the valve is maintained greater than 40 degrees open.

The Nuclear Service Water System will still be capable of performing its accident mitigation functions. Catawba UFSAR Section 9.2.1.3 specifies that the Nuclear Service Water System is designed to supply the cooling water requirements of a simultaneous LOCA on one unit and cool down on the other unit assuming a single failure anywhere on the system, loss of offsite power and loss of Lake Wylie. Upon complete channel separation, both units are assured of having a source of water, at least one pump capable of supplying required flow on its associated channel, and at least one essential header to provide cooling water to components served by the Nuclear Service Water System. The LOCA event is the only event identified which has radiological consequences. The contributors leading up to overall consequences are identified in UFSAR 15.6.5.3 for a LOCA. Each of these contributors were reviewed to determine if they are affected by the partial stroke and removal of valve 1RN-003A.

The partial stroke and removal of valve 1RN-003A does not affect the release of fission products during a LOCA. The Nuclear Service Water System supports the Component Cooling Water System, which provides Reactor Coolant Pumps seal integrity. Alterations to the Nuclear Service Water System alignment do not affect the support to the Component Cooling Water System. The partial stroke and removal of valve 1RN-003A does not affect the ability of the Nuclear Service Water system to provide assured makeup to the Containment Valve Injection Water System. The Annulus Ventilation system and concrete containment wall is not affected by the partial stroke and removal of valve 1RN-003A. The partial stroke and removal of valve 1RN-003A does not affect leakage rates of components outside of containment. The removal systems (ice condenser, containment spray, radioactive decay) are not affected by the partial stroke and removal of valve 1RN-003A. Since none of the individual contributors are affected then the partial stroke and removal of valve 1RN-003A does not affect radiological consequences.

The Nuclear Service Water System will still be capable of performing its accident mitigation functions. The Nuclear Service Water alignment in this temporary procedure will still result in the Nuclear Service Water System operating within its design basis. No new credible failure modes are associated with the partial stroke and removal of valve

1RN-003A. The possible equipment malfunctions are identified in the Safety Review above and each malfunction is bounded in the UFSAR. None of the malfunctions identified result in increased radiological consequences.

There are no new credible failure modes associated with this new temporary procedure. The partial stroke and removal of valve 1RN-003A does not change the analyzed operating conditions for the Nuclear Service Water System and therefore does not create the possibility of a different type of accident nor does it increase the probability of an accident previously considered not credible.

The equipment malfunction scenarios identified in the Safety Review above bounds the malfunction of equipment from these proposed changes. All of these malfunctions are within the bounds of equipment malfunctions identified in the SAR. The Nuclear Service Water System will still operate within its design basis. No new failure modes beyond those currently analyzed in the SAR are created. The Nuclear Service Water system will still be capable of performing its accident mitigation functions. No accident or equipment malfunction previously thought not credible has been made credible.

There is no unreviewed safety question associated with this activity. No Technical Specification changes are required. No UFSAR changes are required.

184 **Type:** Miscellaneous Items **Unit:** 0

Title: Temporary Station Modification CNTM-0055, Block Nuclear Service Water System Pit Emergency Low Pit level signals to Nuclear Service Water Pumps and Train B Crossover Valves

Description: Temporary Station Modification CNTM-0055 will block the Nuclear Service Water pump start signals and the Nuclear Service Water crossover valve close signals when a Nuclear Service Water Pit Emergency Lo level signal is initiated.

During Unit 1 Refueling Outage EOC12, both trains of the Nuclear Service Water System will be aligned to the Standby Nuclear Service Water Pond (SNSWP), the Nuclear Service Water System return header crossover valves 1RN-54A and 1RN-53B will be tagged closed in their safe position, and 120VAC Power Panelboard 1ERPA will be de-energized for approximately 12 hours for maintenance. When 1 ERPA is de-energized the Nuclear Service Water System Pit A, 2-out-of-3, Emergency Low Level swap logic signal will be initiated. An Emergency Low Level Pit signal on one Pit also initiates the opposite Pit's logic. Therefore, both train's valves and the Nuclear Service Water System Pumps receive swap signals. Temporary Station Modification CNTM-0055 opens sliding links to block the Nuclear Service Water System Pit Emergency Lo level signal to start each the Nuclear Service Water System Pump and opens sliding links to block the Nuclear Service Water System crossover valve close signals. The Modification Test Plan ensures the proper links are opened and restored.

Evaluation: The Nuclear Service Water System including Lake Wylie and the SNSWP is the ultimate heat sink for various nuclear safety related heat loads during normal operation and design basis events. The Nuclear Service Water System also supplies emergency makeup water to various nuclear safety related systems during design basis events.

Catawba UFSAR Section 9.2.1.3 states:

The Nuclear Service Water System is designed to supply the cooling water requirements of a simultaneous LOCA on one unit and cooldown on the other unit assuming a single failure anywhere on the system, loss of offsite power and loss of Lake Wylie. Upon complete channel separation, both units are assured of having a source of water, at least one pump capable of supplying required flow on its associated channel, and at least one essential header to provide cooling water to components served by the Nuclear Service Water System.

Prior to blocking the Nuclear Service Water System Pit Emergency Low Level signal to each Nuclear Service Water System Pump, the Nuclear Service Water System will be aligned to the SNSWP per Operations procedure OP/0/A/6400/006C, "Nuclear Service Water System." Temporary Station Modification CNTM-0055 will block the auto start of Nuclear Service Water System Pumps 1A, 2A, 1B, and 2B and closing of the Nuclear Service Water System supply header crossover valves 1RN48B, 2RN48B, 1RN47A, and 2RN47A when a Nuclear Service Water System Pit Emergency Low Level signal is initiated. This is acceptable since the failure of one Nuclear Service Water System train to swap to the SNSWP is no longer credible and train separation upon receipt of a pit swap signal is no longer required. Pump operator controls from the Control Room and Auxiliary Shutdown Panels and automatic pumps starts on Safety Injection and Loss of

Offsite Power will not be affected by this Temporary Station Modification. The ability to provide train separation will be available because Nuclear Service Water System crossover valve operator controls and automatic closure due to Containment Hi-Hi Pressure will not be affected by this Temporary Station Modification. In this configuration, both trains of the Nuclear Service Water System will be operable. Therefore the actions associated with Technical Specification 3.7.8 are not applicable. Technical Specification 3.3.2, ESF Table 3.3.2-1 requires 3 channels operable per pit in Modes 1-4. The conditions required for one inoperable channel is to place the channel in trip or align to the SNSWP. The condition required for two inoperable channels is to align to the SNSWP. The Nuclear Service Water System will be aligned to the SNSWP prior to blocking the Nuclear Service Water System Pit Emergency Low Level logic to the Nuclear Service Water System Pumps, therefore, Technical Specification conditions will be satisfied.

There is no unreviewed safety question associated with this procedure revision. No Technical Specification changes are required. No UFSAR changes are required.

174 Type: Miscellaneous Items **Unit:** 1

Title: Temporary Station Modification Work Order 98205724 (01/02) Bypass P-12 Interlock in Mode 4 for Extended Cooldown on Condenser Steam Dump Valves

Description: Temporary Station Modification (TSM) Work Order 98205724 (01/02) provides the method for bypass of the P-12 interlock and provides a method to use additional condenser steam dump valves for unit cooldown while in procedure OP/1/A/6100/002. The P-12 interlock will be bypassed in the Auxiliary Safeguards Cabinets to disable the interlock when appropriate pressure and temperature conditions are met during Unit 1 cooldown in Mode 4. Technical Specification 3.3.2 requires that the interlock be operable during Modes 1, 2, and 3. This interlock may be bypassed when the unit is in Mode 4 since it is no longer required by Technical Specifications. The condenser steam dumps are controlled using the Steam Pressure Controller before and after the P- 12 interlock is bypassed. This controller can be operated in auto with a steam pressure setpoint or in manual with a pushbutton demand signal. This procedure reduces the amount of time the Residual Heat Removal System is needed to operate during unit cooldown by performing an extended cooldown using condenser dump valves at lower temperatures. This method of cooldown is expected to reduce the amount of crud precipitated upon start of the Residual Heat Removal System and lower general area dose rates during shutdown.

There are two major issues to be considered for this change.

1. The ability to add positive reactivity at a faster rate than would be possible using only three steam dump to condenser valves at which the TSM is installed (290-300 deg F).

Procedure OP/1/A/6100/002 has been revised to include provisions for either Mode 6 boron concentration or the boron concentration associated with the final temperature of the cooldown prior to utilizing the TSM to make available all three banks of steam dump to condenser valves for cooldown. Thus, adequate shutdown margin will be maintained and return to criticality is not possible.

2. The ability to cool the Reactor Coolant System and potentially challenge the Technical Specification cooldown limit curve for Unit 1 given in Technical Specification Figure 3.4.3-2 will be afforded by the six additional dump valves' heat removal capability.

An evaluation was performed to assess the cooldown potential following failure of the steam dump controller after placing the TSM in service (when Reactor Coolant System Tavg is 300 degrees F or below). It was determined that the Technical Specification cooldown limit of 100 degrees F/hour should not be violated due to this failure alone with all nine steam dump to condenser valves open. It was also shown that an existing failure mode of the Residual Heat Removal System flow control valve failing open would lead to a cooldown rate more severe than a failure of the steam dump controller at these temperatures/pressures. Failure of the steam dump controller at Reactor Coolant System temperatures just below the P-12 setpoint (553 degrees F) and the associated opening of just one bank (Bank #1) of steam dump to condenser valves results in a much worse cooldown by comparison. Thus, Pressurizer Thermal Shock events will not be exacerbated by this alternate cooldown method.

Evaluation: There are no Unreviewed Safety Questions associated with the use of TSM Work Order

98205724. No Technical Specification changes are required. Various sections of the UFSAR could be revised to clarify that alternative methods of cooldown are available and may be used at temperatures below 300 degrees F. However, until this evolution is deemed to be a permanent change a UFSAR change will not be made. If it is decided the UFSAR should reflect the alternative method of cooldown using the condenser dump valves (requiring the P-12 interlock bypass) and delaying placing the Residual Heat Removal System in operation later than currently described in the UFSAR, a UFSAR change will be prepared at that time.

248 **Type:** Miscellaneous Items **Unit:** 0

Title: Update to the SIMULATE-3P/TABLES-3 software

Description: The SIMULATE-3P/TABLES-3 software is being updated. The current software, SIMULATE-3P Version 4 is being upgraded to SIMULATE-3P Version 6, and TABLES-3 Version 4 is being upgrade to TABLES-3 Version 5. A benchmark evaluation was performed to show that these two SIMULATE-3P software versions are computationally and functionally equivalent. Catawba Nuclear Design will begin implementation of the new SIMULATE-3P software version for core designs beginning with Catawba Unit 2 Cycle 12. This software and the workstations used to run it are not part of any System, Structure, or Component as described in the UFSAR.

Evaluation: SIMULATE-3P is an advanced two-group nodal code based on the QPANDA neutronics model. This code is used for the calculation of steady state and transient power distributions, control rod worths, integral reactivity worths, reactivity coefficients, and kinetics data. TABLES-3 is the data processing code that links CASMO-3 cross section data to SIMULATE-3. The CASMO-3/SIMULATE-3P code package is the NRC approved methodology used by Duke Power for calculating nuclear physics parameters per topical report DPC-NE-1004A. SIMULATE-3P Version 6 and TABLES-3 Version 5 were certified in accordance with Duke Power Nuclear System Directive NSD-800 and verified to yield the same results as SIMULATE-3P Version 4 and TABLES-3 Version 4 except for the modifications described below. These enhancements inconsequentially affected SIMULATE-3P's nuclear physics parameter calculation.

SIMULATE-3P Version 6 incorporates the following enhancements:

Modification to calculate the 2-D exposure by using a weighted (by segment loading) collapse of 3-D exposures.This will correctly calculate the 2-D exposure for assemblies with annular axial blankets.
Enhanced CPU utilization.
Increased default iteration and enhanced convergence such that a tighter converged solution will result.
A new ASME steam table option is available which improves handling of cross section data because the pre-built, tabled ASME steam table option used is more rigorous than the previous version. For "off nominal" thermal and hydraulic conditions analyzed in Steam Line Break Accident the resulting power peaking is slightly more conservative due to the steam table nodal cross section treatment for saturated nodes.
Improved output/error edit information for QV&V.
SIMULATE-3P Version 6 is the current vendor supported SIMULATE-3P version.
Other miscellaneous modifications to increase the usability of the code.

McGuire and Catawba SAR references to this software are:

DPC-NF-2010A "Nuclear Physics Methodology for Reload Design",
DPC-NE-1004A "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P",
DPC-NE-2011PA "Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors",
DPC-NE-2009PA "Westinghouse Fuel Transition Report",
DPC-NE-3001-PA "Multidimensional Reactor Transients and Safety Analysis Physics

Parameters Methodology",
DPC-NE-3002-PA "FSAR Chapter 15 System Transient Analysis Methodology",
DPC-NE-2012A "Dynamic Rod Worth Measurement Using CASMO/SIMULATE",
DPC-NE-1003-A "Rod Swap Methodology Report for Startup Physics Testing".
UFSAR Sections 4.1, 4.2, 4.3, 4.4, and Chapter 15 discuss SIMULATE-3P calculations performed by this software and references the above SAR documents.
Technical Specification 5.6.5 and Bases list these SIMULATE-3P methods as an approved COLR method.

The SIMULATE-3P calculations discussed in these SAR documents are unchanged.

The SIMULATE-3P software, and workstations where it resides, are not part of any SSC important to safety and do not directly affect any SSC's. The software is not installed at the plants, but rather on workstations in the Nuclear General Office. Regulatory commitments associated with this software have been reviewed and determined acceptable. A reactivity management review was performed and no concerns were identified.

SIMULATE-3P Version 6 was verified to yield the same results as SIMULATE-3P Version 4, except for the new enhancements. The modifications were verified in accordance with NSD800 and are in compliance with Technical Specifications and approved methods. No unreviewed methods are introduced by this code revision. Based on the foregoing discussion, this change does not adversely affect any design bases, safety functions, safety limits, safety margins, setpoints, or core parameters. The capability to shutdown the plant and maintain it in a safe condition is unaffected. There are no unreviewed safety questions associated with this software upgrade. No Technical Specification changes are required. No changes to the UFSAR are required.

113 **Type:** Miscellaneous Items **Unit:** 1

Title: Valve 1NI181 Reseating Troubleshooting Plan

Description: Pressure boundary valve (PBV) leakage and Residual Heat Removal System pressurization are requiring frequent system venting by Operations. In an effort to reseat the secondary pressure boundary check valves known to be leaking, a troubleshooting sequence used in Operations Procedure OP/0/A/6350/014, Operations Troubleshooting Guidelines, has been developed that directs the operator to open the bypass line around the primary pressure boundary valve on the affected cold leg. Opening the bypass isolation valve establishes the maximum amount of differential pressure across the ten-inch cold leg accumulator (CLA) check valve, the six-inch Residual Heat Removal System secondary PBV, and the two-inch Safety Injection System secondary PBV.

Evaluation: Primary bypass line isolation valves 1(2)NI-391, 1(2)NI-392, 1(2)NI-393, and 1(2)NI-394) are three quarter inch air-operated, globe valves controlled from the Safety Injection System test panel located in the Auxiliary Building. The valves are normally closed with control power removed during plant Modes 1-4. Each valve fails closed on loss of air or loss of control power to the solenoid. In addition, the valve fails closed when plant operation is transferred to the Standby Shutdown Facility (SSF). Position indication for the valves is not affected by loss of control power or SSF operation. If one of these valves fails to close when required during the performance of this procedure, the operator is instructed to remove control power by using the key switch at the Safety Injection System test panel.

It is important to distinguish between a valve that fails to close and one that fails to indicate closed. The primary bypass isolation valves depend on limit switches for open and closed indication at the Safety Injection System test panel. A number of problems with limit switch indication have been encountered during outage PBV testing. The problem seen most often is that both open and closed indication for a valve is lost when the valve is stroked. Troubleshooting this kind of problem usually finds that the valve is stroking open and closed, but one or both of the limit switches are not being picked up. The design of these valves makes it highly unlikely that one would fail to close if required. The Safety Injection System test panel valves have very little travel between full open and closed. These valves are safety related and are air to open and if power or air is lost, an internal spring will cause the valve to close. Should it become necessary to close any particular primary bypass valve during the performance of this procedure, the fact that the valve would be operated against no differential pressure provides additional assurance of valve closure regardless of limit switch indication.

Opening a primary bypass isolation valve will cause pressure to equalize across the first ten-inch PBV off the Reactor Coolant System resulting in a differential pressure of almost 2200 psig across the secondary check valves back to the Safety Injection System and the Residual Heat Removal System. The CLA secondary check valve will have almost 1600 psid. Differential pressures of at least 1400 psid are needed to obtain essentially leak tight performance from these valves due to their wide seat width. This same alignment is performed in Mode 4 at a Reactor Coolant System pressure of approximately 1300 psig during initial PBV testing prior to startup from refueling. Piping downstream of (and including) each secondary PBV is Duke Class A and rated to withstand full Reactor Coolant System temperature and pressure. Piping sections upstream of each secondary

PBV, though rated for full Reactor Coolant System pressure back through and including their motor operated isolation valves, are protected by ASME code relief valves with somewhat lower set pressures. The relief valves are in place to relieve over pressure conditions associated with limited leakage past any of the secondary PBV's. This procedure directs the operator to immediately close any primary bypass isolation valve, should Safety Injection System/Residual Heat Removal System pressure or CLA level begin increasing in an uncontrolled manner.

The allowable leakage past pressure boundary check valves is limited by Technical Specification 3.4.14. PBV's are required to close with limited leakage to prevent diversion of reactor coolant to the Safety Injection System or Residual Heat Removal System. Also, the limited leakage provides additional assurance of valve integrity, thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Exposing the secondary pressure boundary valves to Reactor Coolant System pressure will not result in leakage in excess of Technical Specification requirements. Each of these valves was tested during the last refueling outage and leakage was verified to be less than the allowable when the test data was corrected to full Reactor Coolant System pressure. Allowing the primary check valve to float with no differential pressure does not invalidate its last test because the valve is not moving off its seat. There is no force present which could cause the primary check valves to pass flow in the forward direction. Flow in the forward direction is the only condition other than damage to a check valve that would invalidate a previously successful PBV test. Should leakage be estimated to exceed that allowed by Tech Spec 3.4.14, the unit would have to comply with the action statement for the applicable condition, isolate the leak, and proceed to a shutdown condition.

No valves other than the primary bypass isolation valves are being manipulated by this procedure. The procedure directs the operator at the Safety Injection System test panel to close any valve that may spuriously open. Double isolation is provided by normally closed manual valves in series with each air-operated valve on all but a few lines controlled from the Safety Injection System test panel. The exceptions are the CLA drain valves and the primary bypass isolation valves described above. Spurious opening or a failure to close of any CLA drain valve would pressurize the Safety Injection System accumulator fill header to CLA pressure. The header is designed to withstand full Reactor Coolant System pressure and is equipped with redundant train related containment isolation valves that close on an ESF signal, therefore this particular failure is of no concern.

Accident operation of any system that serves an ECCS function is unaffected by the performance of this procedure. Should an accident occur while the primary bypass isolation valve is open, the operator is directed by the procedure to return the valve to the closed position. If the operator is unable to perform this task for some reason, air supply to the valve would still be lost due to an accident-generated Hi-Hi Containment pressure (Sp) signal that isolates the Instrument Air supply to containment. As previously described, each valve fails closed on loss of air. In any case no ECCS flow would be directed away from the core. The bypass line connects back into ten inch CLA cold leg injection line and the delay time to the core for that portion of the flow is negligible.

There is no unreviewed safety question associated with this modification. No Technical Specification changes are required. No UFSAR changes are required.

82 **Type:** Miscellaneous Items **Unit:** 2

Title: Work Orders 98269585 and 98269945 describing use of the Reactor Building Polar Crane to replace hydrogen igniters

Description: Work Orders 98269585 and 98269945 are a maintenance activity. Glow plugs of both Train A and B of the Unit 2 Hydrogen Mitigation System were replaced. This activity required movement of the polar crane to position a person under the glow plug to be replaced. The crane operators followed the safe load path that was developed for movement of the polar crane while the associated Unit (in this case, Unit 2) is in Modes 1, 2, 3, or 4. In addition, the breaker for the main hook was open during the operation. These precautions precluded the possibility of a load drop on any system, structure, or component important to safety and also ensured compliance with NUREG 0162 (Phase 1).

Evaluation: There is no unreviewed safety question associated with the activity. The activity has no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. No UFSAR change is required.

26 Type: Nuclear Station Modification **Unit:** 0

Title: CN-50447/00, Relocate the Nuclear Service Water Pumphouse Ventilation System Fan Selector Switches from the Control Room to the Nuclear Serrvice Water Pump Structure, Delete associated Auxiliary Shutdown Panel wiring and add a low temperature annunciator.

Description: Nuclear Station Modification CN-50447/00 will modify the controls for the Nuclear Service Water Pump Structure Ventilation System.
New local (to the pump structure) controls are being added and Control Room controls are being deleted. A Nuclear Service Water Pump Structure selector switch, for each fan, with "Auto-Off positions will be added in each pumphouse. Control Room Annunciators are being modified to indicate conditions of Lo and Hi pumphouse temperature. Additionally, Auxiliary Shutdown Panel controls and wiring will be deleted as it is no longer necessary.

Problems with the existing design include:

1) Control problems related to power bleed over into the control circuitry caused interaction between the Auxiliary Shutdown Panel wiring and large power cables routed nearby.

2) Concerns associated with Nuclear Service Water Pump Structure Ventilation System failures that can lead to low temperatures within the Nuclear Service Water Pumphouse.

Some cables are being deleted due to eliminating the Control Room and Auxiliary Shutdown Panel controls. The reason the Auxiliary Shutdown Panel - Nuclear Service Water Pump Structure Ventilation System control interface exists is so that following a fire in the Control Room which may result in fire-induced failures, the Nuclear Service Water Pump Structure Ventilation System control can be resumed upon transfer at the Auxiliary Shutdown Panel. Eliminating the Control Room controls makes the Auxiliary Shutdown Panel controls unnecessary. The Nuclear Service Water Pump Structure Ventilation System will operate automatically based on the need for cooling and/or the presence of an Engineered Safety Features Actuation Signal. The local controls at the Nuclear Service Water Pump Structure involve wiring within each separate pumphouse. Therefore, no Appendix R concerns have been created. Post modification testing will ensure that the modified system performs as anticipated.

Evaluation: No new failures are introduced as a result of this modification. This modification will make the Nuclear Service Water Pump Structure Ventilation System more reliable. The revised design offers more protection to the existing failure involving outside air intake during cold weather (failure of dampers to go to recirculation alignment). The "Lo" temperature Control Room annunciator will provide indication at 45 degrees F. if no earlier observations are made. The control board has not been degraded due to the elimination of the Nuclear Service Water Pump Structure Ventilation System controls. No Unreviewed Safety Questions are created by this modification. No Technical Specification changes are required. Changes are required to UFSAR Sections 7.6.21 and 9.4.8.

131 **Type:** Nuclear Station Modification **Unit:** 1

Title: Nuclear Station Modification CN-11351/00 Digital Rod Position System Upgrade

Description: Nuclear Station Modification CN-11351/00 will replace the single Digital Rod Position Indication (DRPI) System display with two separate and independent computer nodes and two CRT displays. The two new computer nodes will perform the data processing functions which were previously performed by the current DRPI display. Also, Operator Aid Computer software will be provided that will allow it to provide rod position information. These changes will essentially give three options of rod position information (two DRPI, one OAC) which could be used to meet Technical Specification surveillance requirements for those Technical Specifications requiring rod position determination (TS 3.1.4, 3.1.7). The power supply arrangement for the DRPI System will also be reconfigured. During a Loss of Off-Site Power (LOOP) Event, the current DRPI System will not indicate rod position due to loss of power since both data cabinets are powered from their respective units RPA/RPB. The current power source for the Unit 1 DRPI display is fed from panelboard 1RPA with 1RPB as the alternate source provided through a transfer switch. These panelboards are fed from a non-essential and non-battery backed power supply. Power will be reconfigured such that Unit 1A data cabinet and display will be powered from Unit 1 RPA and Unit 1B Data Cabinet will be powered from Unit 1 RPB. Additionally Unit 1B Data cabinet and display will have an alternate power supply via an automatic transfer switch from the Unit 2 RPB. This represents a change in power supply for the B Data Cabinet in that there is a backup power supply and it comes from the opposite Unit. Each separate DRPI system (A/B data cabinet and display) will have a separate power source and the B display and data cabinet has an automatic backup from the other Unit. This makes DRPI more reliable for Loss of Power (LOOP) scenarios involving Unit 1. The susceptability of losing all DRPI from a Unit 1 Loop has been removed. Half accuracy will be retained on Unit 1 from a Unit 1 LOOP since B Data cabinet and display will automatically swap to Unit 2 RPB. Additional redundancy is provided by the separate computer nodes which will improve reliability of DRPI from card and other component related failures. Also, as long as the data cabinets and one computer node are operating, the OAC can provide full accuarcy rod position indication. Loss of a single power supply to a data cabinet will still result in half accuracy DRPI indication as it did prior to this modification. The power for the Unit 2 DRPI System will be affected by this modification as described above. This is apparent since 2RPB can be powering Unit 1 DRPI equipment (data cabinet and display) in emergency backup mode and also be used as primary power for the Unit 2 DRPI B data cabinet simultaneously. Following implementation of modification CN-21351, the power arrangements will be similar to the Unit 1 configuration with the 2B data cabinet and display backup getting power from the opposite unit (1RPB).

Evaluation: This modification has no effect on the probability or consequences of accidents evaluated in the UFSAR. The DRPI system is not nuclear safety related, does not perform any accident mitigation function, and is not an accident initiator. The ability to comply with Technical Specifications is not degraded and will actually be improved due to better fault-tolerance and redundancy of the system. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. UFSAR Section 3.1 and 7.7.1.3.2 will be revised.

99 **Type:** Nuclear Station Modification **Unit:** 1

Title: Nuclear Station Modification CN-11385/00, Addition of Bonnet Vents on Valves 1NI-121A and 1NI-152B and New Actuator on Valve 1NV-252A to eliminate pressure locking concerns

Description: Modification CN-11385/00 modifies valves 1NI-121A, 1NI-152B, and 1NV-252A to prevent pressure locking as committed in response to NRC Generic Letter 95-07.

Valves 1NI-121A and 1NI-152B are the isolation valves between the Safety Injection Pump Dscharge and the Reactor Coolant System Hot Legs. The valves also serve as outside containment isolation valves. The valves are normally closed with power removed and remain closed during post-LOCA safety injection and cold leg recirculation. When necessary, power is restored and the valves are opened to establish hot leg recirculation. The Containment Valve Injection Water System supply to the valves will be deleted and, for each valve, the Containment Valve Injection Water System connection to the valve bonnet will be used to install a one-half inch vent path to the hot leg side of the valve. Removing the Containment Valve Injection Water System connection was discussed with NRC personnel and its justification is documented in calculation CNC-1223.12.000062, Justification for Removal of Containment Valve Injection Water System Supply from Valves NI121A and NI152B. The calculation shows that, even without the Containment Valve Injection Water System, containment atmosphere will be isolated at these penetrations (M317 and M320) in the event of a Design Basis Accident. During safety injection and cold leg recirculation, Safety Injection System pressure against the closed valves will be greater than Containment pressure. During hot leg recirculation, the valves are open and passing flow into Containment. The worst case would be during a Safety Injection System pump failure. Its associated valve (NI-121 or NI-152) would not be opened for hot leg recirculation and have only Residual Heat Removal discharge pressure against the closed valve. Even in this case, pressure will still be greater than Containment pressure. The applicable Safety Injection System design conditions of stainless steel, Duke Class B, 2500 psia/650 degrees F are still met for the added piping and valves. The vent line piping and components were reviewed and found acceptable in the areas of pipe rupture, stress analysis and seismic support.

Valve 1NV-252A is the normally closed motor-operated gate valve that opens to provide flow from the Refueling Water Storage Tank to the Centrifugal Charging Pumps during accident conditions. The actuator will be replaced with a heavier Rotork brand actuator, but will still operate at the same speed; hence, the valve stroke time is unaltered. The effects of the heavier actuator on the existing valve were evaluated by Westinghouse and documented in manufacturer's drawing CNM 1205.19-0103.001, 09D-235 Seismic and Weak Link Analysis of Westinghouse 8-in Gate Valve. The analysis was performed for deadweight plus pressure plus seismic plus operating loads. Catawba Engineering has analyzed the proposed configuration and found no additional seismic supports or restraints are required. The only electrical equipment change is the replacement of overload heaters located in motor control center 1EMXA.

No changes are being made to the ECCS actuation circuitry. No electrical power or control changes are part of this modification. The piping stress analysis and support/restraint designs have been evaluated for these changes. The ability of these

valves to respond to accident conditions is not degraded. No accident input assumptions are invalidated; therefore, the consequences of design basis accidents evaluated in the UFSAR are unaffected. The required design specifications of seismic integrity, pressure/temperature limits, material selection, ASME code class, etc. are maintained.

Evaluation: There are no unreviewed safety questions associated with this modification. No Technical Specification changes are requiried. USFAR Table 6-77 (pg 5) and its Notes are being revised to show that valves 1NI-121A and 1NI-152B no longer receive Containment Valve Injection Water System injection and the justification for not performing a leak rate test. UFSAR Table 3-104 (pg 26) will be revised to remove the Containment Valve Injection Water System supply valves (1NW-190A and 1INW-232B) to the Safety Injection System valves from the active valve list.

UFSAR Figure 6-116 will be revised to remove the Containment Valve Injection Water System supply to valves 1NI-121A and 1NI-152B. UFSAR Figure 6-130 will be revised to include the bonnet vent paths on valves 1NI-121A and 1NI-152B.

78 Type: Nuclear Station Modification **Unit:** 1

Title: Nuclear Station Modification CN-11393/00 Main Feedwater System Containment Isolation Valve Actuator Modification

Description: Modification CN-11393/00 will resolve problems associated with the Unit 1 Feedwater System Isolation valves (1CF-33, 42, 51, and 60). These valves are 18 inch Borg-Warrner pneumatic-hydraulic operated gate valves. This modification adds a nuclear safety related nitrogen accumulator and associated tubing/fittings/isolation valves to increase design basis closing margin for the Main Feedwater Isolation Valves (MFIVs). It also removes the nitrogen solenoid valves, to eliminate chronic leakage problems, and replaces the existing hydraulic cylinder end cap which is necessary to facilitate solenoid valve removal. This modification also replaces a nitrogen pressure switch with a pressure transmitter to provide a more reliable method of nitrogen pressure indication along with analog inputs for OAC alarms. Finally, the filters and orifices on the hydraulic solenoid valves will be replaced with transfer tubes to improve performance.

Also, the flow diagram (CN-1591-1.1 Rev 25A) is being modified to delete Note 1. This note stated that a stroke time of five seconds was required for valves 1CF-33, 42, 51, and 60. License Amendment No. 107 (Unit 1) changed the 5 second stroke time to "N/A". Therefore, this change is only modifying the drawing to make it match the as analyzed and approved (by NRC) plant.

Evaluation: The MFIVs are considered "equipment important to safety" in that they perform both a containment isolation function and the Feedwater Isolation "Engineered Safety Features" function. This modification will not degrade the MFIVs' capability to respond to any of its applicable actuation signals. The additional nitrogen bottle is added to improve design basis closing margin. The quality and classification of the added components (nitrogen tank, tubing, valves, and instrumentation) is consistent with the existing classification of components so no degradation is imposed. Also, the environmental qualification of the MFIVs is not degraded with respect to the existing components. The ability to maintain adequate nitrogen pressure will be improved with respect to standby readiness with the analog output of the new transmitter. No new power requirements are involved with this modification. New cables are required for the transmitter. No Appendix R concerns were identified. All applicable design criteria have been preserved in this design.

Consideration was given to making the valve actuator more reliable through less leakage (pneumatic solenoid valve removal) and providing more design basis closing margin. These changes have been accomplished while maintaining a nuclear safety related, single failure proof design which can isolate the MFIVs following a safety related closure signal assuming the single failure of any component. A failure modes and effects was completed to justify pneumatic solenoid valve removal. The solenoid valve arrangement was changed but the removal of the pneumatic solenoid valves actually removes a potential failure (opening of the solenoid valves on demand). In summary, no new failure modes were created and a potential failure was eliminated. No common mode failures are introduced as nuclear safety related devices are being added and no adverse interactions are created since seismic mounting is provided for nuclear safety related equipment.

The relevant accidents for the MFIVs are those resulting in the applicable actuation signals: Safety Injection, S/G hi- hi level, Rx Trip with lo-Tavg, and Doghouse Water

level hi-hi all result in closure signals to some or all the MFIVs. Significant credit for feedwater isolation occurs in the accidents discussed in the design basis document; Excessive Feedwater Accident Due to Feedwater Control Valve Failure (UFSAR 15.1.2) and Steam Line Rupture (UFSAR 15.1.5). Other accidents resulting in any of the listed signals (e.g. LOCA) will also result in MFIV closure. The consequences of these accidents could only be affected if the performance of the valves was altered such that the valves would no longer meet acceptance criteria. As discussed above, this modification will result in increased design basis closing margin. All assumptions in accident analyses continue to be fulfilled as assumed. The solenoid valve arrangement is changing but the removal of the pneumatic solenoid valves actually removes a potential failure (opening of the solenoid valves on demand). The stroke time requirement for the valves is not changing as well as the actuation signals to the remaining hydraulic solenoid valves. The performance of the valves is not degraded and the stroke time will be tested after modification installation. While the solenoid valves between the MFIV and the Nitrogen tanks have been removed, to eliminate a leakage component, a prior evaluation has determined that the MFIV is not more likely to fail closed, even though the actuator is aligned directly the nitrogen tanks.

No unreviewed safety questions are created by modification CN-11393/00. No Technical Specification changes are required. UFSAR Figure 10-28, a system flow diagram will be revised.

133 **Type:** Nuclear Station Modification **Unit:** 1

Title: Nuclear Station Modification CN-11394/00, Upper Internals Guide Tube Support (Split Pin) Replacement Project

Description: Modification CN-11394/00 will replace the upper internal guide tube support pins, which are known as split pins. Stress corrosion cracking of these split pins has occurred in the industry. These failures are not a nuclear safety issue, only an economic one. Loose parts due to failures have been discovered at other Westinghouse plants. The currently installed split pins are fabricated from Inconel X-750 Rev. A and each pin has the appearance of a "clothes pin". A split pin has two separate legs that have an interference fit with a hole in the upper core plate. The split pins are attached to the bottom flange of a guide tube assembly via its shank, nut and locking device. The pins are captured with a washer welded to the bottom of the guide tube support. The material for the replacement split pin is SA-316CW which is qualified and approved for this application. The modification to the upper internals include the following activities:

1.Replace all split pins (a quantity of 146) with cold worked type 316 stainless steel material.

2.Remove all guide tube/flow column hold down bolts (a quantity of 536) and install approximately 50% (4 vs. 8) of the hold down bolts fabricated from cold worked type 316 stainless steel material, except for the guide tubes adjacent to the outlet nozzles. The guide tubes located adjacent to the outlet nozzles will have all 8 replacement bolts installed.

3.Remove the 15 x 15 guide tube flexures (a quantity of 24) and removable inserts (a quantity of 6) and replace them with mechanical plugs.

4.Remove the small orifice cover plates (a quantity of 4) from 17 x 17 plutonium recycle guide tubes

5.Remove the large orifice cover plates (a quantity of 8) from the 17 x 17 guide tubes and replace with like material.

Note: Items 2 through 5 have to be removed in order to access the split pins to effect replacement. It is cheaper and easier to install new bolts (item 2) than to reuse existing bolts. The flexures (item 3) have caused problems in some plants so removal and replacement with plugs is desirable. Item 4 was possible due to item 3 plug installation. Item 5 is a remove/replace exercise with like material.

Evaluation: An evaluation was performed, WCAP-15252 Rev 1, to document the acceptability of the replacement CW316 support pin material and pin design. The engineering evaluation documents the acceptability of the pin and nut material and assembly. In so doing, WCAP-15252 Rev 1 documents the functional equivalency of the replacement support pin and nut assembly relative to the previous components. Also, the functional equivalency of the associated replacement hardware such as the guide tube hold down cap screws, guide tube cover plate hardware (which are essentially like-in-kind replacements), is provided. The reactor components being modified are nuclear safety related components. The split pin modification is a nuclear safety related modification. The replacement split pins are procured as nuclear safety related material.

There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. No UFSAR changes are required.

79 Type: Nuclear Station Modification **Unit:** 1

Title: Nuclear Station Modification CN-11396/00 Install new eight inch Motor Operated Valves (1RN-250A and 1RN-310B)

Description: Modification CN-11396/00 installs new 1RN-250A and 1RN-310B valves (Nuclear Service Water System to Auxiliary Feedwater System boundary valves). Instead of placing the valves in the current 1RN-250A, 1RN310B location, the new valves are being located as close as possible to the 24-inch Nuclear Service Water System supply header. Placing the valves close to the Nuclear Service Water System header minimizes the volume of stagnant lake water in the to Auxiliary Feedwater System piping thus minimizing corrosion and fouling concerns. The new valves are 8-inch motor-operated gate valves.

New 2-inch drain valves, 1RNE-97 and 1RNE-98, are being added upstream of the new boundary valves. The existing 6-inch 1RN-250A/1RN-310B valves will be renamed 1CA-300/1CA-308 and electrically disconnected along with being locked open. All other valves and flow instrumentation between the new boundary valves and 1CA-300/1CA-308 will be renamed as part of the Auxiliary Feedwater System and required to meet the to Auxiliary Feedwater System design conditions.

In addition to the valve work, the 6-inch Nuclear Service Water System to Auxiliary Feedwater System piping will be replaced from the Nuclear Service Water System supply header to just upstream of the Auxiliary Feedwater System flow-measuring orifice with 8-inch pipe to reduce pipe friction losses. The new 8-inch pipe between the 24-in Nuclear Service Water supply headers and the new RN-250A/RN-310B valves will be mostly a superaustenitic stainless steel that is very resistant to corrosion (a very short section will be carbon steel from the Nuclear Service Water System supply header to a new flange to which the superaustenitic steel will be connected).

Evaluation: The new valves and pipe and their configuration have been evaluated as acceptable for the pressure, temperature, weight, thermal and seismic loads. Stress and support analyses do not reveal any adverse conditions. The new valves and pipe has been evaluated as acceptable for the installed location. There are no pipe rupture interaction concerns created. There are no high energy or moderate energy concerns because the pipe routing is the same general area with no susceptible equipment along the routes. The new valves and associated cabling were evaluated and cable routing design criteria was satisfied. Electrical separation criteria and Appendix R requirements were applied.

Modification CN-11396/00 does not involve an unreviewed safety question. No Technical Specification changes are required. USFAR Table 1-4 is being revised to include a new Auxiliary Feedwater System flow diagram to relieve the overcrowding caused by this modification on an existing flow diagram. UFSAR Table 3-104 and Table 7-15 are being revised to show the larger valve size for 1RN-250A and 1RN-310B and to indicate the revised ESF response time. UFSAR Figures 9-27, 9-28, 9-31 and 10-33 (system flow diagrams) will be revised.

98 **Type:** Nuclear Station Modification **Unit:** 1

Title: Nuclear Station Modification CN-11398/00, Modify "C" Heater Drain Pump Controls to trip on Turbine Trip

Description: Modification CN-11398/00 will modify the control circuitry for the "C Heater Drain Tank Pumps" such that both 1C1 and 1C2 pumps trip when the Main Turbine is in a tripped condition. Also, procedure OP/2/A/6100/03 will be revised to ensure "C Heater Drain Pumps" are secured below 70% power. This will allow one of the conditions of operability associated with an "Operable but Degraded" evaluation to be made permanent. Another modification, CE-10831, will isolate valves 1(2)CM-33 and 1(2)CM-127. The scope of this modification relates only to adding the "C Heater Drain Pumps" trip on Turbine Trip and a permanent procedure change to secure these pumps below 70% power (when they are not needed).

The prohibition against "C Heater Drain Pump" operation below 70% helps minimize the potential for the "C Heater Drain Pumps" to dead-head the Condensate Booster Pumps over a large operating range for which they are not needed. Dead-heading the Condensate Booster Pumps can lead to a trip at the 3000 gpm setpoint. Tripping the Condensate Booster Pumps will lead to tripping the Main Feedwater System pumps and the actuation of the Auxiliary Feedwater Pumps for decay heat removal. This modification will help keep the Main Feedwater system operating in recirculation mode to the condenser, following transients like a Reactor or Turbine Trip. This will allow for easier plant start-up and post trip recovery efforts. This item also adds additional margin against the potential for overpressurization of Condensate System piping since valve CM-127 is isolated.

It has been determined that the secondary systems at Catawba could have contained the design feature to Trip the "C Heater Drain Tank Pumps" when the Main Turbine is tripped in the original design. Tripping these pumps does not have any degrading effects on plant response. Based on this evaluation, the design features contained in the systems are adequate to avoid a Main Feedwater System pump trip on low suction flow and pressure in particular for Turbine Trip that occurs from conditions below the P9 setpoint (69 %) where an automatic Reactor Trip will not occur. The load rejection valve, CM-83, along with starting idle Hotwell and Condensate Booster Pumps for various conditions involving lost margin is adequate to preclude tripping the Main Feedwater System pumps. Thus, this modification does not increase the likelihood of Reactor Trips for Turbine Trips below P9 and challenges to the Auxiliary Feedwater System, both of which could occur if the Main Feedwater System pumps were to trip, post Turbine Trip. The Condensate and Feedwater Systems will continue to perform design functions without any degrading effects due to tripping the "C Heater Drain Pumps" on Turbine Trip and the continued prohibition against "C Heater Drain Pump" operation below 70%.

Evaluation: There are no unreviewed safety questions associated with modification CN-11398/00. No Technical Specification changes are required. Another modification, CE-10831, will isolate both valves 1(2)CM-33 and 1(2)CM-127. UFSAR changes will be processed then to show valves 1CM-127 and 1CM-33 isolated. No changes to the UFSAR are required for modification CN-11398/00.

126 **Type:** Nuclear Station Modification **Unit:** 1

Title: Nuclear Station Modification CN-11399/00

Description: Modification CN-11399/00 will implement changes to the Condensate Systems that are related to the reliability of Auxiliary Feedwater System suction sources. Valve CM-33 will have a control switch added that will allow the operators to take manual control of the valve to open, or close, or place the valve in automatic. Also, an automatic action is being added to valve CS-47 to close on an Auxiliary Feedwater System automatic start signal along with a reset button. Valves CS-33 and CS-57 will have a note on the flow diagram that shows them as normally isolated whenever valve CM-127 is isolated to prevent a loss of inventory from the Auxiliary Feedwater System. In addition a new hotwell level instrument will be added with an extended range.

Valve CM-33 is an automatic valve that opens and aligns a flowpath from the hotwell to the Upper Surge Tank on high discharge pressure from the hotwell pumps or high level in the hotwell. This valve automatically controls hotwell level during power maneuvers by opening to allow the hotwell pumps to deliver excess hotwell inventory to the Auxiliary Feedwater System. It also provides a hotwell pump minimum flow path.

Even though this modification is providing a control switch for manual operation of valve CM-33, the ability to use this flowpath is dependent on unisolating the flowpath to valve CM-33 by opening valve CM-32 and its bypass valve CM-146 shown on UFSAR Figure 10-18. Another modification currently relies on the path through valve CM-35 for manual make-up. A future change (either modification or procedure change) will be necessary to utilize the path through valve CM-33 by opening valve CM-32 and its bypass valve CM-146.

Manual valves CS-32 and CS-35 are contained in the emergency make-up line from the Upper Surge Tank to the hotwell and valve CS-32 isolates automatic valves CS-33 and CS-57. Valves CS-32 and CS-35 will have a note on the flow diagram that shows them as normally closed whenever valve CM-127 is isolated to prevent a loss of Auxiliary Feedwater System inventory from the Upper Surge Tank. The emergency make-up lines serve as a recirculation path back to the hotwell when valve CM-127 is routing flow to the Upper Surge Tank. By closing valves CS-32 and CS-35, the emergency make-up line is effectively isolated. This action provides further assurance that the Upper Surge Tank, as a suction source for the Auxiliary Feedwater System, is preserved and protected from certain failures (non-safety failures of Air Operated Valves CS-33 and CS-57 closure) resulting in Upper Surge Tank depletion. Normal Hotwell make-up on lo-level is still available from the Upper Surge Tank via valve CS-47 discussed above. Thus, the unit is still able to tolerate condensate transients that reduce Hotwell level within the capability of the CS-47 flowpath. Any challenges beyond this are deemed to result in a Auxiliary Feedwater System actuation anyway (e.g. Feedwater Line Break).

The new control switch for valve CM-33 and reset switches, one for each train of Auxiliary Feedwater System automatic start, for valve CS-47 have been qualified for addition to the Control Board 1MC-13. No Appendix R concerns are created. The Safe Shutdown capability of the plant is not degraded by this modification. No degradation has been imposed on any power supplies. No control circuit problems have been created.

Thus, no new failure modes are created. The interface between the Auxiliary Feedwater System automatic start signal and controls for valve 1CS-47 utilizes adequate electrical separation between the safety and non-safety circuits so as not to degrade the safety related circuits. The changes made by this modification will improve the reliability of the Auxiliary Feedwater System by isolation of valves CS-33/ CS-57 and automatic closure of valve CS-47 on Auxiliary Feedwater System automatic start. Both of these changes preserve Upper Surge Tank inventory, which is preferred as a suction source over the hotwell.

Evaluation: No unreviewed safety questions are created by modification CN-11399/00. No Technical Specification changes are required. UFSAR changes are required for Section 10.4.1.5 and 10.4.7.5.2 to describe the isolation of valves 1CS-33 and 1CS-57. UFSAR Figures 9-59 (Condensate System flow diagram) will be revised accordingly. Also, changes to section 10.4.7.5.2 are required to describe the manual control switch for valve 1CM-33.

128 **Type:** Nuclear Station Modification **Unit:** 1

Title: Nuclear Station Modification CN-11400/00, Additional Temperature Monitoring in Upper Surge Tank and Hotwell, and New Control Switches to Block Auxiliary Feedwater System Pump Low Suction Pressure Trip

Description: Modification CN-11400/00 will implement changes to the Condensate Systems that are related to the reliability of Auxiliary Feedwater System suction sources. This modification will install new temperature instrumentation at the top of each Upper Surge Tank and provide more accurate indication of temperature near the point where recirculation from the Condensate Booster Pump discharge (Valve CM-127) is added. Also, new temperature instrumentation will be added in the condenser hotwell. Additionally, control switches will be added in the Control Room that will allow the Auxiliary Feedwater Pumps low suction pressure trip to be blocked.

Currently, the blocking of the Auxiliary Feedwater Pump low suction pressure trip is performed in the field by sliding links. The new Control Room switches will provide a more convenient and less intrusive method of blocking the trip function for better utilization of the hotwell that does not require field work or entering cabinets.

This modification involves three new Control Board switches (CA-5, CA-6, and CA-11) for which the board is qualified . There will be a switch for A Train Motor Driven Auxiliary Feedwater Pump, B Train Motor Driven Auxiliary Feedwater Pump, and the Turbine Driven Auxiliary Feedwater Pump.

UFSAR Section 10.4.9.2 "Auxiliary Feedwater System" and Section 7.4. 1 "Auxiliary Feedwater System Instrumentation and Control" were reviewed. The low suction pressure pump trip function when in manual control is not described in the text. It appears on UFSAR Figures 7-6 and 7-7 as a "stop pump" function and "do not align with Nuclear Service Water System" function if certain conditions exist which include manual Auxiliary Feedwater System operation and no Auxiliary Feedwater System auto-start signal.

Currently, emergency procedures, AM/1/A/5100/001, provides for defeating the low suction pressure Auxiliary Feedwater Pump trip by sliding the appropriate links. Thus, the defeat switch is not providing any capability that is not currently available. This action is only taken after resetting the Auxiliary Feedwater System following an Auxiliary Feedwater System autostart. When manual control is taken, the Nuclear Service Water System auto swap circuitry is disabled unless a valid Auxiliary Feedwater System autostart signal is generated. Also, the low suction pressure pump trip is enabled unless defeated through some means (sliding links per procedure or via switches post mod). UFSAR Figures 7-6 and 7-7 illustrate the requirements for alignment to the Nuclear Service Water Pond.

The emergency state of the plant is deemed to have been terminated when manual control is taken. Any subsequent plant conditions resulting in another Auxiliary Feedwater System autostart, such as blackout or lo-lo Steam Generator levels, would enable the Nuclear Service Water System autoswap circuitry and if the pressure setpoints are satisfied, a swap to the Nuclear Service Water System would occur. Thus, the defeat of

the pump trip does not impair subsequently swapping to the Nuclear Service Water System, if conditions indicate the need, which maintains this Engineered Safety Feature. This modification neither blocks an ESF (Auxiliary Feedwater System Autostart) nor prevents an ESF (autoswap to Nuclear Service Water System) from being initiated if needed.

The additional temperature monitoring is an improvement with respect to Auxiliary Feedwater System reliability in that better utilization of Condensate sources for Auxiliary Feedwater System will be provided. The Auxiliary Feedwater temperature requirement of 138 degrees F. identified in UFSAR 10.4.9.2 can be monitored more effectively with the new temperature monitoring arrangement.

This modification installs new temperature instrumentation in the Upper Surge Tank approximately one foot from the top of the tank. This should eliminate the requirement for the Upper Surge Tank to be overflowing to the Condensate Storage Tank to prevent temperature errors since the hot water that may be of concern for Auxiliary Feedwater System operability would come into the top of the tank first. Additionally, new temperature instrumentation will be installed in the condenser hotwell approximately one foot up from the bottom of the hotwell. The accuracy of the new instrumentation will be sufficient to allow for a 136 degree F. setpoint to assure water no greater than 138 degree F. is present . The Control Room will utilize an annunciator to alarm undesirable temperature conditions in the Upper Surge Tank, Condensate Storage Tank, or hotwell.

Evaluation: No power supplies are degraded by any load changes involved with this modification. Protective devices (fuses) are adequately sized . No Appendix R concerns are created. The Safe Shutdown capability of the plant is not degraded by this modification . The temperature instrumentation will be installed with quality consistent with the systems, structures, and components to which they interface. The new Control Room switches will provide a more convenient and less intrusive method of blocking the trip function for better utilization of the hotwell. No degradation is imposed on the Auxiliary Feedwater System, or the Condensate Systems. The response of the Auxiliary Feedwater System to any Chapter 15 accident analysis for which the system is intended to respond, is unchanged by this modification. The limiting accident analysis assumes the Condensate Storage System is not available since it is non-safety related and credits proper operation of the Nuclear Service Water System for non-SBO events.

There are no Unreviewed Safety Questions associated with this modification. No Technical Specification changes are necessary. UFSAR Table 10-18 and Figure 9-59 will be revised.

27 **Type:** Nuclear Station Modification **Unit:** 1

Title: Nuclear Station Modification CN-11401/00, Reroute and Enlarge Upper Surge Tank to Auxiliary Feedwater Condensate Storage Tank Piping

Description: Nuclear Station Modification CN-11401/00 will modify non-nuclear safety related piping and valves into a more reliable Condensate Storage System. The modification will reroute and enlarge Upper Surge Tank to Auxiliary Feedwater Condensate Storage Tank piping with the goal of increasing flow through the Upper Surge Tank piping. This modification will enlarge the piping connecting to the Auxiliary Feedwater Condensate Storage Tank and lower the elevation of the "tee" where the two flowpaths intersect. Approximately 450 feet of 8 inch piping will be replaced with 12 inch piping from the Upper Surge Tank to the junction between the Auxiliary Feedwater Condensate Storage Tank and Upper Surge Tank. The Upper Surge Tank to Auxiliary Feedwater Condensate Storage Tank junction tee will be lowered approximately 13 feet. Fewer elbows will be used to reduce friction. Valve 1CA-004 (Auxiliary Feedwater Pump Suction from Upper Surge Tank Isolation Valve) and valve 1CA-006 (Auxiliary Feedwater Pump Suction from to Auxiliary Feedwater Condensate Storage Tank Isolation Valve) will be relocated for easier access for manual operation from the 568' elevation. Valve 1CA-003 will be replaced with a new 12 inch check valve and valve 1CA-004 will be replaced with a new 12 inch motor operated valve. Valve 1CS-019 will be replaced with a new 16 inch butterfly valve. Power and control cables will be moved for valves 1CA-004 and 1CA-006. One inch vent and drain globe valves 1CA-013 and 1CA-195 are being replaced with the same size ball valves. New one inch vent and drain valves 1CS-119, 1CS-120, 1CS-121 are being added.

Electrical scope of the modification includes power to the new motor operated valve for 1CA-004, which is the same as the existing motor operator for 1CA-004 . Power and control cables for valves 1CA-004 and 1CA-006 will be relocated. Valves 1CA-004 and 1CA-006 are powered from the Blackout Power System to ensure the valves may be closed whenever the contents of the Auxiliary Feedwater Condensate Storage Tank are depleted. No increase in electrical load is being placed on the Blackout Power System. Protective devices are adequately sized. No Appendix R concerns are created. The Safe Shutdown capability of the plant is not degraded by this modification. The additional suction head and flow benefits derived from this modification will improve the reliability of the Auxiliary Feedwater System with respect to undesirable swaps to the Nuclear Service Water System prior to full usage of the Upper Surge Tank and (following modifications CE-61531 and CE-61532, which reopens valves 1/2 CA-006) the Auxiliary Feedwater Condensate Storage Tank.

Evaluation: There are no unreviewed safety questions associated with this modification. The modification affects non nuclear safety related piping and valves. The modification will result in a more reliable Condensate Storage System. This system is used as an accident mitigation system as a suction source to the Auxilairy Feedwater System. No Technical Specification changes are required. UFSAR Figures 9-59 (Condensate System flow diagram) and 10-33 (Auxiliary Feedwater System flow diagram) are being revised to show the piping and valve changes.

129 **Type:** Nuclear Station Modification **Unit:** 1

Title: Nuclear Station Modification CN-11405/00, Residual Heat Removal/Containment Spray System Sump Pump Control Interlock with SSPS

Description: Modification CN-11405/00 will install an electrical interlock between the Unit 1 SSPS System and both the 1A and 1B Residual Heat Removal System/Containment Spray System sump pump control circuits. The interlock is described in UFSAR Section 6.3.2.5 and SER Section 6.3.2 but was never actually installed at Catawba. The purpose of the interlock is to give the operators early indication of ECCS leakage so that the proper course of action can be taken to mitigate the leakage. The interlock stops (locks-out) the pumps on a Safety Injection (SI) signal. Subsequent leakage into the sump will result in the accumulation of water to the Hi-Hi setpoint. Then a Control Room Operator Aid Computer alarm will alert the operators to the abnormal sump level and at the Hi-Hi setpoint the standby pumps start to remove water. Since the interlock will lock-out the pumps on a Safety Injection (SI) signal, a reset switch will be provided in the control room to return the pumps to their normal status. Also, a new Control Room annunciator will be added to the existing Operator Aid Computer alarm at the Hi-Hi level for the Unit 1 level switches to accommodate "Loss of Operator Aid Computer" scenarios.

Wiring will be installed from the Unit 1 SSPS cabinets to accommodate the Unit 1 interlocks to the Unit 2 Residual Heat Removal System/Containment Spray System sump pumps 2A and 2B. This wiring was not installed under modification CN-21405/0 due to concerns about working in the Unit I SSPS cabinets while Unit 1 was on-line. This circuitry from Unit 1 will interface with the 2A and 2B Residual Heat Removal System/Containment Spray System sump pump control circuitry if necessary, by design.

Modification CN-21405/00 partially implemented the interlocks from Unit 2 SSPS to the Unit 1 Residual Heat Removal System/Containment Spray System sump pumps 1A and 1B to avoid working in the Unit 2 SSPS cabinets during implementation of modification CN-11405/00. CN-11405/00 will complete this change by finishing the wiring from the Unit 2 SSPS cabinets to the Unit 1 2B Residual Heat Removal System/Containment Spray System sump pumps 1A and 1B. Post mod testing for this modification will cycle the slave relay in the Unit 2 SSPS cabinets.

The Residual Heat Removal System/Containment Spray System sump pumps are nuclear safety related and are considered accident mitigation equipment. The interlock being added is described in the UFSAR Section 6.3.2.5 and SER Section 6.3.2 but was never installed at Catawba. The purpose of the interlock is to give the operators early indication of ECCS leakage so that the proper course of action can be taken to mitigate the leakage. Therefore, the proposed activity is making the Residual Heat Removal System/ Containment Spray System sump pump controls match the SAR description. Post accident response for accidents involving ECCS can now be mitigated in accordance with the SAR. No accident analysis assumptions are affected by this modification.

Evaluation: The controls are being modified consistent with standards. In summary, this modification will modify the Residual Heat Removal System/Containment Spray System sump pump controls to make them match the SAR description and no degradation is imposed.

These pumps are support systems for the equipment located in the Residual Heat

Removal System/Containment Spray System pump rooms, on elevation 522' of the Auxiliary Building, in that they prevent flooding.

There are no Unreviewed Safety Questions associated with this modification. No Technical Specification changes are required. No changes to the UFSAR are required.

132 **Type:** Nuclear Station Modification **Unit:** 2

Title: Nuclear Station Modification CN-21386/00, Addition of Bonnet Vents on valves 2NI-121A and 2NI-152B and new actuators on valves 2NS-38B and 2NV-252A to eliminate pressure locking concerns.

Description: Modification CN-21386/00 modifies valves 2NI-121A, 2NI-152B, 2NS-38B and 2NV-252A to prevent pressure locking of these valves as committed in response to Generic Letter 95-07.

2NI-121A and 2NI-152B are the isolation valves between the Safety Injection Pump discharge and the Reactor Coolant System hot legs. Also, these valves serve as outside containment isolation valves. The valves are normally closed with power removed and remain closed during post-LOCA safety injection and cold leg recirculation. When necessary, power is restored and the valves are opened to establish hot leg recirculation. The Containment Valve Injection Water System supply to the valves will be deleted and, for each valve, the Containment Valve Injection Water System connection to the valve bonnet will be used to install a one half inch vent path to the hot leg side of the valve. Removing the Containment Valve Injection Water System connection was discussed with NRC personnel and its justification is documented in calculation CNC-1223.12.000062, "Justification for Removal of Containment Valve Injection Water System Supply from Valves NI-121A and NI-152B". The calculation shows that, even without the Containment Valve Injection Water System, containment atmosphere will be isolated at these penetrations (M317 and M320) in the event of a Design Basis Accident. During safety injection and cold leg recirculation, Safety Injection System pressure against the closed valves will be greater that Containment pressure. During hot leg recirculation, the valves are open and passing flow into Containment. Worst case would be during a safety Injection pump failure. Its associated valve (NI-121 or NI-152) would not be opened for hot leg recirculation and would have only Residual Heat Removal System discharge pressure against the closed valve. Even in this case, pressure will still be greater than Containment pressure. The applicable Safety Injection System design conditions of stainless steel, Duke Class B, 1915 psia/200 degrees F, and 2500 psia/650 degrees F are still met for the added piping and valves. The vent line piping and components were reviewed and found acceptable in the areas of pipe rupture, stress analysis and seismic support.

Valve 2NS-38B is the normally closed motor-operated containment isolation valve that when opened provides flow from Residual Heat Removal Pump B to its respective Residual Heat Removal System Auxiliary Containment Spray Header. The Auxiliary Containment Spray Headers are an additional provision to the Containment Spray System and are manually aligned after swapover to recirculation mode to help complete containment heat removal. Valve 2NS-38B will automatically close per the Containment Pressure Control System to prevent excessive Containment depressurization through inadvertent or excessive operation of the engineered safety features as described in UFSAR Section 7.6.4. In order to provide greater operator margin in opening against a pressurized bonnet, a new actuator is being installed with greater thrust capability. The new actuator is the same size and weight, but the actuator speed is lowered to increase the thrust output. The valve stroke time increases from 7.3 seconds to 9.8 seconds. Slowing the stroke time by 2.5 seconds does not adversely impact the Auxiliary Containment

Spray capability since its service is manually aligned. The electrical characteristics of the new actuator are the same as the existing actuator; therefore, no overload heaters or circuit changes are necessary.

Valve 2NV-252A is the normally closed motor-operated gate valve that opens to provide flow from the Refueling Water Storage Tank to the Centrifugal Charging Pumps during accident conditions. The actuator will be replaced with a heavier Rotork brand actuator, but will still operate at the same speed; hence, the valve stroke time in unaltered. The effect of the heavier actuator on the existing valve was evaluated by Westinghouse and documented in CNM 1205.19-0103.001, 09D-235 Seismic and Weak Link Analysis of Westinghouse 8-in Gate Valve. The analysis was performed for deadweight plus pressure plus seismic plus operating loads. Catawba Engineering has analyzed the proposed configuration and found no additional seismic supports or restraints are required. The only electrical equipment changes necessary is the replacement of the overload heaters located in motor control center 2EMXA.

No changes are being made to the Emergency Core Cooling System actuation circuitry. No electrical power or control changes are part of this modification. The piping stress analysis and support/restraint designs have been evaluated for these changes. The ability of these valves to respond to accident conditions is not degraded. No accident input assumptions are invalidated; therefore, the consequences of design basis accidents evaluated in the UFSAR are unaffected. The required design specifications of seismic integrity, pressure/temperature limits, material selection, ASME code class, are maintained.

Evaluation: Modification CN-21386/00 does not involve an unreviewed safety question. No Technical Specification changes are required. USFAR Table 6-77 (page 29) and its Notes are being revised to show valves 2NI-121A and 2NI-152B no longer receive Containment Valve Injection Water System injection and the justification for not performing a leak rate test. UFSAR Table 3-104 (pages 69 and 70) will be revised to remove the Containment Valve Injection Water System supply valves (2NW-190A and 2NW-232B) to the Safety Injection System valves from the active valve list.
Since Unit 2 flow diagrams are not in the UFSAR, but selected Unit 1 drawings are, the UFSAR flow diagrams will not change until the equivalent Unit 1 modification is implemented.

80 **Type:** Nuclear Station Modification **Unit:** 2

Title: Nuclear Station Modification CN-21396/00 Install new eight inch Motor Operated Valves (2RN-250A and 2RN-310B)

Description: NSM CN-21396/00 installs new 2RN-250A and 2RN-310B valves (Nuclear Service Water System to Auxiliary Feedwater System boundary valves). Instead of placing the valves in the current 2RN-250A, 2RN-310B location, the new valves are being located as close as possible to the 24-inch (Train A) and 20-inch (Train B) Nuclear Service Water System supply header. Placing the valves close to the Nuclear Service Water System header minimizes the volume of stagnant lake water in the to Auxiliary Feedwater System piping thus minimizing corrosion and fouling concerns. The new valves are 8-inch motor-operated gate valves.

New two-inch drain valves, 2RNE-97 and 2RNE-98, are being added upstream of the new boundary valves. The existing 6-inch 2RN-250A/2RN-310B valves will be renamed 2CA-300/2CA-308 and electrically disconnected along with being locked open. All other valves and flow instrumentation between the new boundary valves and 2CA-300/2CA-308 will be renamed as part of the Auxiliary Feedwater System and will be required to meet the to Auxiliary Feedwater System design conditions.

In addition to the valve work, the 6-inch Nuclear Service Water System to Auxiliary Feedwater System piping will be replaced from the Nuclear Service Water System supply header to just upstream of the Auxiliary Feedwater System flow-measuring orifice with 8-Inch pipe to reduce pipe friction losses. The new 8-inch pipe between the Nuclear Service Water System supply headers and the new RN-250A/RN-310B valves will be mostly a superaustenitic stainless steel that is very resistant to corrosion (a very short section will be carbon steel from the Nuclear Service Water System supply header to a new flange to which the superaustenitic steel will be connected).

Evaluation: The new valves and pipe and their configuration have been evaluated as acceptable for the pressure, temperature, weight, thermal and seismic loads. Stress and support analyses do not reveal any adverse conditions. The new valves and pipe has been evaluated as acceptable for the installed location. There are no pipe rupture interaction concerns created. There are no high energy or moderate energy concerns because the pipe routing is the same general area with no susceptible equipment along the routes. The new valves and associated cabling were evaluated and cable routing design criteria was satisfied. Electrical separation criteria and Appendix R requirements were applied.

Modification CN-21396/00 does not involve an unreviewed safety question. No Technical Specification changes are required. UFSAR Table 3-104 and Table 7-15 are being revised to show the larger valve size for 2RN-250A and 2RN-310B and to indicate the revised ESF response time. Other revisions to the USFAR will be made because of the Unit 1 version of this modification (CN11396). USFAR Table 1-4 is being revised to include a new to Auxiliary Feedwater System flow diagram to relieve the overcrowding caused by this modification on an existing flow diagram. UFSAR Figures 9-27, 9-31 and 10-33 (system flow diagrams) are being revised.

54 **Type:** Nuclear Station Modification **Unit:** 2

Title: Nuclear Station Modification CN-21398/00 Modify "C" Heater Drain Pump Controls to trip on Turbine Trip

Description: Nuclear Station Modification CN-21398/00 will modify the control circuitry of the "C" Heater Drain Tank Pumps such that both the 2C1 and 2C2 Pumps trip when the Main Turbine is in a tripped condition. Also, Procedure OP/2/A/6100/03 will be revised to ensure "C Heater Drain Pumps" are secured below 70% power. As such, one of the conditions of operability associated with an "Operable but Degraded" condition from Corrective Action Program Problem Report 0-C98-1726 will be made permanent. This modification will not clear the Operable but Degraded situation. Another modification (CE-10831) will permanently close both valves 1(2)CM-33 and 1(2)CM-127. The scope of the modifcation described herein is only related to adding the "C Heater Drain Pumps" trip on Turbine trip and the procedure change is to secure these pumps below 70% power (when they are not needed).

The prohibition against "C Heater Drain Pump" operation below 70% power helps to minimize the potential for the "C Heater Drain Pumps" to "dead-head" the Condensate Booster Pumps over a large operating range for which they are not needed. Dead-heading the Condensate Booster Pumps can lead to a trip at the 3000 gpm setpoint. Tripping the Condensate Booster Pumps will lead to tripping the Main Feedwater Pumps and ultimately actuation of the Auxiliary Feedwater Pumps for decay heat removal. This modification will help keep the Main Feedwater System operating in the recirculation mode to the Condenser, following transients like a reactor or turbine trip. This will allow for easier plant startup and post trip recovery efforts. This modification also adds additional margin against the potential for overpressurizing the Condensate System piping.

Evaluation: There are no unreviewed safety questions associated with this Nuclear Station Modification. The secondary systems for Catawba Nuclear Station could have included the design feature to trip the "C Heater Drain Tank pumps" when the Main Turbine is tripped in the original design. Tripping these pumps does not degrade plant response. Per this evaluation the design features contained in the systems are adequate to avoid a Main Feedwater Pump trip on low suction flow and pressure in particular for the Turbine Trip that occurs from conditions below the P9 (69% power) setpoint where an automatic reactor trip will not occur. The load rejection valve (CM-83), along with starting idle Hotwell and Condensate Booster Pumps for varoius conditions involving lost margin is adequate to preclude tripping the Main Feedwater Pumps. Therefore the modification does not increase the probability of reactor trips for turbine trips below P-9 and does not present challenges to the Auxiliary Feedwater System, both of which could occur if the Main Feedwater Pumps were to trip after a turbine trip. This change does not degrade the performance of the Condenste/Feedwater Systems. No Technical Specification changes are required. No UFSAR changes are required.

253 **Type:** Nuclear Station Modification **Unit:** 2

Title: Nuclear Station Modification CN-21401/00 Reroute and Enlarge Upper Surge Tank to Auxiliary Feedwater Condensate Storage Tank Piping

Description: Modification CN-21401/00 will modify non-nuclear safety related piping and valves that will result in a more reliable Condensate Storage System. The modification will reroute and enlarge piping from the Upper Surge Tank to the new junction between the Upper Surge Tank and Auxiliary Feedwater Condensate Storage Tank, with the goal of increasing flow through the Upper Surge Tank piping. This modification will also lower the elevation of the "tee" where the two flowpaths intersect. Approximately 450 feet of eight inch pipe will be replaced with twelve inch pipe from the Upper Surge Tank to the junction between the Auxiliary Feedwater Condensate Storage Tank and Upper Surge Tank. There will also be approximately 50 feet of 16 inch piping installed. The Upper Surge Tank to Auxiliary Feedwater Condensate Storage Tank junction tee will be lowered approximately 14 feet. Fewer elbows will be used to reduce friction. Valves 2CA-004 (Auxiliary Feedwater Pump Suction from Upper Surge Tank Isolation Valve) and 2CA-006 (Auxiliary Feedwater Pump Suction from Auxiliary Feedwater Condensate Storage Tank Isolation Valve) will be relocated for easier access for manual operation. Valve 2CA-003 will be replaced with a new 12 inch check valve and valve 2CA-004 will be replaced with a new 12 inch motor operated valve. Valve 2CS-019 will be replaced with a new 16 inch butterfly valve. Power and control cables will be moved for valves 2CA-004 and 2CA-006. One inch vent and drain globe valves 2CA-013 and 2CA-195 are being replaced with the same size ball valves. New one inch vent and drain valves 2CS-126, 2CS-127, 2CS-128 are being added.

The electrical portion of the modification includes power to the new motor operated valve for CA-004, which is the same as the existing motor operator for CA-004. Power and control cables for valves CA-004 and CA-006 will be relocated. Valves 2CA-004 and 2CA-006 are powered from the Blackout Power System to ensure the valves may be closed whenever the contents of the Upper Surge Tank and Auxiliary Feedwater Condensate Storage Tank are depleted. No increase in electrical load is being placed on the Blackout Power System. Protective devices are adequately sized. No Appendix R concerns are created. The Safe Shutdown capability of the plant is not degraded by this modification.

Control Board 2MC-10 is also being modified with the addition of a receiver gauge to display Auxiliary Feedwater Condensate Storage Tank level for the dedicated Unit 2 Auxiliary Feedwater Condensate Storage Tank. The level instrument was added as part of modification CN-50477/0. The control board has been evaluated for the addition of the new gauge.

The additional suction head and flow benefits derived from this modification will improve the reliability of the Auxiliary Feedwater System with respect to undesirable swaps to the Nuclear Service Water System prior to full usage of the Upper Surge Tank and (following modification CNCE-61532, which reopens valves 2CA-006) the Auxiliary Feedwater Condensate Storage Tank.

Evaluation: No unreviewed safety questions are created by this modification. No Technical Specification changes are required. UFSAR Figure 9-59 (Condensate System flow

diagram) and Figure 10-33 (Auxiliary Feedwater System flow diagram) were revised to show the piping and valve changes for the corresponding Unit 1 changes under modification CN11401/00. No other UFSAR changes are required.

53 **Type:** Nuclear Station Modification **Unit:** 2

Title: Nuclear Station Modification CN-21405/00 Residual Heat Removal/Containment Spray Sump Pump Control Interlock with SSPS

Description: Nuclear Station Modification CN-21405/00 will install an electrical interlock between the Solid State Protection System (SSPS) and both of the Residual Heat Removal/ Containment Spray sump pump control circuits. The interlock is described in UFSAR Section 6.3.2.5 and SER Section 6.3.2 but was never installed at Catawba. The purpose of this interlock is to give the operators early indication of Emergency Core Cooling System (ECCS) leakage so that steps can be taken to mitigate the leakage. The interlock stops (locks-out) the pumps on a safety injection signal. Subsequent leakage into the sump will result in the accumulation of water to the Hi-Hi setpoint. Then a Control Room Operator Aid Computer (OAC) alarm will alert the operators to the abnormal sump level and at the Hi-Hi setpoint the standby pumps start to remove water. Since the interlock will lock-out the pumps on a safety injection signal, a reset switch will be provided in the Control Room to return the pumps to their normal status. Also a new Control Room annunciator will be added to the existing OAC alarm at the Hi-Hi level to accommodate "Loss of OAC" scenarios.

Evaluation: There is no unreviewed safety question associated with this nuclear station modification. These sump pumps are nuclear safety related and are considered accident mitigation equipment. Installation of this interlock will bring the plant into compliance with the UFSAR and the SER. No Technical Specification changes are required. No UFSAR changes are required.

81 **Type:** Nuclear Station Modification **Unit:** 0

Title: Nuclear Station Modification CN-50477/00 Filtered Water Storage Tank Conversion to Unit 2 CACST and existing shared CACST conversion to Unit 1 CACST

Description: Modification CN-50477/00 is modifying non-nuclear safety related piping and valves that will result in a more reliable Condensate Storage System. Specifically, Filtered Water Storage Tank "C" will be converted to a Unit 2 Auxiliary Feedwater Condensate Storage Tank (CACST) and the existing shared CACST will be refurbished into a Unit 1 specific CACST. Following modification CN-50477/0, unit specific CACST(s) will have the capability of being aligned to the Auxiliary Feedwater pumps suction, resulting in a more versatile and reliable design. Other modifications, CE-61531 and CE-61532, will make the actual valve alignment by opening the currently closed CACST isolation valves, 1CA-6/2 CA-6. Modification CN-50477/00 is providing the equipment for this conversion.

Originally the Filtered Water System consisted of four Filtered Water Storage Tanks. The fourth tank was converted to the existing shared CACST. This is being done once again. The removal from service of the "C" Filtered Water Storage Tank does not affect any systems important to safety. This is reflected in the safety evaluation from UFSAR Section 9.2.4.3.

Evaluation: The systems, structures and components added by this modification are of consistent quality and specification as the existing plant equipment to which they interface. The design temperature and pressures are compatible. The Filtered Water System is not important to safety. The Condensate Storage System is important to safety, in that it provides a redundant and diverse Auxiliary Feedwater System suction source, if available, that would otherwise place additional burden on the safety related Nuclear Service Water System. The reliability of the Condensate Storage System has been improved with the dedicated CACSTs. The limiting accident analysis assumes the Condensate Storage System is not available since it is non-safety related and credits proper operation of the Nuclear Service Water System. The Auxiliary Feedwater System response for some design events, which are not UFSAR Chapter 15 events, may be improved by the additional volume provided via the dedicated CACSTs. No Appendix R concerns are created by the electrical scope of this modification.

The additional dedicated volume of condensate can provide margin to meeting the 3.7.6 LCO requirement of 225,000 gallons once valves 1CA-2/2CA-2 are reopened under modifications CE-61531 and CE-61532. The basis to Tech Spec 3.7.6 implies it is necessary to meet the Station Blackout requirement since no AC power is available; thus the Nuclear Service Water System is not available for the four hour coping period. The Tech Spec basis for 3.7.6 could be revised to allow for the CACST volumes to be used to meet the LCO requirement of 225,000 gallons when modifications CE-61531 and CE61532 are implemented.

No unreviewed safety questions are created by modification CN-50477/00. No Technical Specification changes are required. UFSAR changes are required to sections 9.2.6.3, 10.4.9.2, Tables 9-11, 9-14, Figures 9-51, 9-61, 10-22, and 10-33.

173 **Type:** Nuclear Station Modification **Unit:** 0

Title: Nuclear Station Modification CN-50492, Nuclear Service Water System Chemical Treatment

Description: Modification CN-50492/00 will implement changes to the Nuclear Service Water System intended to improve reliability of the system by minimizing degradation via general corrosion, Microbiologically Influenced Corrosion (MIC), localized pitting due to galvanic attack, and fouling caused by biological slimes and aquatic life along with deposition of silt and corrosion products on the pipe and component walls. Chemically treating the Nuclear Service Water System by injecting a dispersant and biocide into the Nuclear Service Water System will prevent or at least reduce fouling and piping/component degradation due to corrosion and biological attack. The effect of the presence of these chemicals (a dispersant and Stabrex, hypobromous acid HOBr - oxidizing biocide) on the Nuclear Service Water System and interfacing system components has been evaluated. It was concluded that the planned chemical treatment will not have any measurable adverse effects on system materials.

The equipment necessary to inject dispersant and biocide into the Nuclear Service Water System will be Vendor Supplied and consist of the following components:

One insulated and heat traced dispersant tank, two insulated and heat traced biocide tanks, two dispersant pumps , two biocide pumps, and one mixing tee.

The vendor will provide all interconnecting piping, valves and instrumentation, all skid mounted. Dispersant and biocide piping will be insulated and heat traced by Duke. A remote shutdown control for the biocide pumps will be provided by the vendor. This function will be accomplished by a radio signal from the Low Pressure Service Water discharge outfall to the skid mounted injection equipment located outside of the Nuclear Service Water System pumphouse.

The mechanical piping and valves provided to transfer the chemicals from the tanks to the Nuclear Service Water System pits interfaces with the Interior Fire Protection System and with the Drinking Water System. Core drills will be necessary for the injection lines and electrical power cables to penetrate the Nuclear Service Water System pumphouse. Electrical power will be required for heat tracing for the injection lines, eyewash stations, chemical injection skid pumps, heat tracing for the eyewash station and remote control at the Nuclear Service Water System discharge. The power being supplied comes from retail lines in the area.

All of the vendor supplied equipment is non-nuclear safety related. The tie-ins between the nonseismic piping and the seismic piping via the fire protection hose rack is qualified by the locked throttled position of seismic boundary valves 1RN-840 and 1RN-842. Additionally, the Nuclear Service Water System has been evaluated for the flow loss associated with the dilution flow for accident conditions including a seismically induced non-nuclear safety related piping failure. The Nuclear Service Water System tie in vent/drain isolation valves for monitoring equipment have been added. Valves 1RNF52 and 1RNF53 have been added and are seismically designed. No power supplies are degraded by any load changes involved with this modification. Protective devices (fuses)

are adequately sized. No Appendix R concerns are created. The Safe Shutdown capability of the plant is not degraded by this modification . Interfaces with the Drinking Water System and Fire Protection System impose no degradation on those systems. The fire protection capability of the plant is not degraded.

Evaluation: No Unreviewed Safety Questions are created by this modification. UFSAR changes are required to sections 9.2.1.2.3, 9.2.1.3, and 9.2.1.6. . UFSAR Figure 9-22 , Figure 9-24, Figure 9-208 (System Flow Diagrams) and Figure 9-141 (System Flow Diagram) will be changed. No Technical Specification changes are required.

227 **Type:** Procedure **Unit:** 0

Title: Procedure IP/0/A/3112/001C, "Installation and Removal of Nuclear Service Water System Low Level Swap Logic Jumpers"

Description: Procedure IP/0/A/3112/001C provides instructions for disabling the Nuclear Service Water System Pit A or B, 2-out-of-3, Emergency Lo Level swap signals by placing jumpers to block the Nuclear Service Water Pit swap logic. The procedure verifies proper jumper installation by independent verification and by opening a sliding link to ensure the jumper is installed properly. Procedure IP/0/A/3112/001C is used in conjunction with Operations procedures OP/0/A/6400/006C, "Nuclear Service Water System" and OP/0/A/6400/006M, "Nuclear Service Water Unwatering Procedure." OP/0/A/6400/006M provides instructions for draining the pits.

Evaluation: The Nuclear Service Water System including Lake Wylie and the Standby Nuclear Service Water Pond (SNSWP) is the ultimate heat sink for various nuclear safety related heat loads during normal operation and design basis events. The Nuclear Service Water System also supplies emergency makeup water to various nuclear safety related systems during design basis events.

Catawba UFSAR Section 9.2.1.3 states:

The Nuclear Service Water System is designed to supply the cooling water requirements of a simultaneous LOCA on one unit and cooldown on the other unit assuming a single failure anywhere on the system, loss of offsite power, and loss of Lake Wylie. Upon complete channel separation, both units are assured of having a source of water, at least one pump capable of supplying required flow on its associated channel, and at least one essential header to provide cooling water to components served by the Nuclear Service Water System.

Prior to blocking a Nuclear Service Water Pit 2-out-of-3 swap logic per procedure IP/0/A/3112/001 C, the Nuclear Service Water System will be aligned to the SNSWP per Operations procedure OP/0/A/6400/006C, "Nuclear Service Water System" and the applicable Nuclear Service Water Pump breakers will be racked out per Operations procedure OP/0/A/6400/006M, "Nuclear Service Water Unwatering Procedure." Installing the jumper will block that pit's 2-out-of-3 swap logic (i.e. swap logic signal to that train's Nuclear Service Water valves, pumps, and opposite train's crossover valves). The opposite pit's 2-out-of3 swap logic is not affected by the jumper installation. Therefore, if a swap signal is initiated from the opposite pit; that pit's Nuclear Service Water valves will receive a swap signal; that pit's Nuclear Service Water Pumps will start; and the opposite train (pit with jumper installed) crossover valves will close. The non racked out Nuclear Service Water Pumps are not affected by the jumper installation and will start on: Safety Injection, Loss of Offsite Power, and Operator Action. In this configuration, one train of Nuclear Service Water will be inoperable. Therefore the actions associated with Technical Specification 3.7.8 are applicable. Technical Specification 3.3.2, ESF Table 3.3.2-1 requires three channels operable per pit in Modes 1-4. The conditions required for one inoperable channel is to place the channel in trip or align to the SNSWP. The condition required for two inoperable channels is to align to the SNSWP. Nuclear Service Water will be aligned to the SNSWP prior to blocking the 2-out-of-3 logic, therefore, Technical Specification conditions will be satisfied. There is no

unreviewed safety question associated with this procedure revision. No Technical Specification changes are required. No UFSAR changes are required

155 **Type:** Procedure **Unit:** 0

Title: Procedure MP/0/A/7150/072, "Main Steam Safety Valve Setpoint Verification Test"

Description: This procedure provides a method for verification/ adjustment of Main Steam Safety Valve (MSSV) setpoint. The physical test process for MSSV setpoint verification is unchanged by this revision as only editorial changes were incorporated from operating experience to increase the clarity of instruction and provide additional documentation of satisfactory completion of the testing activity and technical specification notifications.

Evaluation: The Main Steam Safety Valves (MSSV's) provide overpressure protection for the secondary system and also provide for overpressure protection of the reactor coolant pressure boundary by providing the heat sink if the preferred heat sink provided by the Ccondenser and the Condenser Circulating Water System is not available. The twenty MSSV's provide relief capacity > 120 % of the rated steam flow and have defined setpoints with a tolerance of +/- 3% required by the Technical Specifications.

This procedure provides a method for verification and adjustment of Main Steam Safety Valve (MSSV) setpoints. The Maintenance procedure was developed from Manufacturer's Instruction Manual CNM-1205.05-5 and from engineering review for the Catawba specific application. Instructions are provided for maintaining the valves within their original design specifications and the instructions are consistent with the industry standard for testing and adjustment this equipment. The actual test has been successfully performed over 1000 times at Catawba and at least 10,000 times across the industry.

The maintenance procedure changes introduced from this revision do not change the test process or methodology used from the previously approved instruction. The changes under review are considered editorial in nature providing increased clarity of instruction and documentation of proper execution within guidelines of station directives. Since the procedure changes addressed by this review are editorial in nature and do not affect the actual test process or methodology, MSSV performance including operating characteristics, failure modes and effects, and all other key elements of safety valve performance will not be affected.

There is no unreviewed safety question associated with this procedure revision. No Technical Specification changes are required. No UFSAR changes are required.

74 **Type:** Procedure **Unit:** 0

Title: Procedure MP/0/A/7450/086A, Revision 0, Control Room Ventilation Ductwork and AHU Access

Description: Procedure MP/0/A/7450/086A Revision 0 "Control Room Ventilation Ductwork and Air Handling Unit (AHU) Access," is a new maintenance procedure that replaces procedure MP/0/A/7450/086 which was inadvertently deleted. The procedure provides guidance for maintenance activities that require access into control room ductwork or air handling units. The procedure recognizes six sections of the ductwork of the system and supplies guidance for maintaining pressure boundary integrity to each section.

Evaluation: The Control Room Ventilation System is a nuclear safety related system whose purpose is to (1) ensure that the control room remains habitable for Operations personnel during and following all credible accident conditions, and (2) ensure that the ambient air temperature does not exceed the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by the system. The system consists of two 100% redundant trains of equipment. The design functions can be adversely affected if the pressure boundary between the redundant trains of equipment is not properly maintained during maintenance activities. The purpose of this procedure is to ensure that steps are taken to properly isolate the two trains or provide a means of restoring any degraded pressure boundary within a time required for proper system operation. The procedure divides the control room ventilation system ductwork into six sections and supplies guidance for maintaining pressure boundary integrity to each section. The procedure contains adequate guidance to ensure that the opposite train of the control room ventilation system is operable and in operation prior to opening any ductwork on the train in which maintenance is to be performed.

The control room ventilation system is not considered an accident initiator. Procedure MP/0/A/7450/086A ensures that one train of the system will be operable and that maintenance on one train does not impact the other train. There are no unreviewed safety questions associated with this procedure. No Technical Specification changes are required. No UFSAR changes are required.

212 **Type:** Procedure **Unit:** 0

Title: Procedure MP/0/A/7650/125A Revision 0, "Removal /Installation of Boot Seals and Penetrations M-394, M-395, and M-371

Description: Maintenance procedure MP/O/A/7650/125 A, revision 0 is a new procedure that will be used during refueling outages while the unit is in Mode 5, 6, or No Mode. During this time the probability and consequences of design bases accidents are low due to the pressure and temperature limitations in Modes 5, 6, and No Mode. The primary issues involved with the temporary conditions in place for outage support are containment integrity for core alterations, depressurization due to a tornado/wind event, and fire barrier integrity.

A qualification for using RTV Silicone Foam as an equivalent method for sealing penetrations was performed and documented in calculation DPC-1435.00-00-0001. The results of this evaluation indicated that the foam seal is acceptable to 17 psi with a cure time of 30 minutes. Technical Specification 3.9.3 "Containment Penetrations" does not prohibit the use of an alternate means to provide containment isolation during core alterations. Catawba Site Directive 3.1.30 defines containment closure as a functional barrier to fission product release under existing plant conditions. The materials used for the pressure boundary seal (bulkhead flanges, foam, hoses, aluminum pipe, pipe caps, and OZ Gedney plugs, etc.) are to prevent free flow of air or communication of the environments between containment and the outside. These non-nuclear safety related materials have been proven through testing to have adequately prevented free air flow and are therefore, not required to be nuclear safety related materials. The use of the bulkhead flanges and the RTV Silicone Foam will not pose any significant additional risk. Likewise the removal of a section of the Boot Seal during Modes 5, 6 and No Mode to support outage work is acceptable with the specified compensatory actions for a tornado/wind event.

The Reactor Building Wall penetration fire barrier and tornado/wind event pressure barrier will be assured by using the RTV foam in conjunction with the fire retardant hoses or pipe through the penetrations. The procedure contains the appropriate prerequisites to verify and notify operations, the containment closure coordinator, and the Senior Reactor Operator to ensure the work is properly sequenced.

Evaluation: There are no Unreviewed Safety Questions associated with procedure MP/O/A/7650/125A. These proposed changes will not create any new types of accidents or malfunctions which are not currently evaluated in the UFSAR. The margin of safety as defined in the bases to the Technical Specifications will not be reduced by these changes. No Technical Specification changes are required. No UFSAR changes are required.

171 **Type:** Procedure **Unit:** 0

Title: Procedure MP/0/A/7650/125A, Revision 001, Removal/Installation of Boot Seal and Penetrations M-394, M-395 and M-371.

Description: Maintenance procedure MP/O/A/7650/125 A, revision 0 is being initiated. This procedure is for use during refueling outages while the unit is in Mode 5, 6, or No Mode. It will provide instructions for temporary bulkhead flange installation and removal, equipment hatch boot seal removal and restoration of a portion of the boot seal in support of Ice Condenser work during refueling outages. During this time the probability and consequences of design bases accidents are low due to the pressure and temperature limitations in Modes 5, 6, and No Mode. The primary issues involved with the temporary conditions in place for outage support are containment integrity for core alterations, depressurization due to a tornado or wind event, and fire barrier integrity.

A qualification for using this RTV Silicone Foam as the equivalent method for sealing penetrations was performed and documented in an engineering calculation. The results of this evaluation indicated that the foam seal is acceptable to 17 psi with a cure time of 30 minutes. The Technical Specification 3.9.3 "Containment Penetrations" does not prohibit the use of an alternate means to provide containment isolation during core alterations. Also a Catawba Site Directive defines "containment closure" as a "functional" barrier to fission product release under existing plant conditions. The materials used for the pressure boundary seal (bulkhead flanges, foam, hoses, aluminum pipe, pipe caps, and OZ Gedney plugs, etc.) are to prevent free flow of air or communication of the containment environment with the outside environment. These non-nuclear safety related materials have been proven through testing to have adequately prevented free air flow and are therefore not required to be nuclear safety related materials. The use of the bulkhead flanges and the RTV Silicone Foam will not pose any significant additional risk. Likewise the removal of a section of the Boot Seal during Modes 5, 6 and No Mode to support outage work is acceptable with the specified compensatory actions for a tornado or wind event in place.

The Reactor Building Wall penetration fire barrier and tornado or wind event pressure barrier will be assured by using the RTV foam in conjunction with the fire retardant hoses or pipe through the penetrations. The procedure contains the appropriate prerequisites to verify and notify operations, the containment closure coordinator, and the SRO to ensure the work is properly sequenced.

Evaluation: There are no Unreviewed Safety Questions associated with this procedure. No Technical Specification changes are required. No UFSAR changes are required.

68 Type: Procedure **Unit:** 0

Title: Procedure OP/0/A/6400/006C, Nuclear Service Water System, Change 224H

Description: The Nuclear Service Water System including Lake Wylie and the Standby Nuclear Service Water Pond (SNSWP), is the ultimate heat sink for various nuclear safety related heat loads during normal operation and design basis events. The Nuclear Service Water System supports Emergency Core Heat Removal (ECCS) operation by providing cooling to the Component Cooling System via the Component Cooling heat exchangers and also to the Diesel Generators (D/Gs) via the D/G Engine Jacket Water Cooler System heat exchangers. Other nuclear safety related loads include the Containment Spray heat exchanger and Control Room Chiller Condenser. The Nuclear Service Water System also provides assured makeup to the Component Cooling System, Spent Fuel Pool, Auxiliary Feedwater supply and the Containment Seal Water Injection System.

The Nuclear Service Water System shall be capable of mitigating the consequences of a design basis event on one unit concurrent with a loss of offsite power on both units, assuming a single failure. The most stringent design basis event for the Nuclear Service Water System is a Loss of Coolant Accident (LOCA) on one unit. The following assumptions are postulated for the Nuclear Service Water design basis LOCA:

Normal Shutdown of the remaining unit from normal operation
Loss of Non-Emergency A/C Power (loss of offsite power) affecting both units
Prolonged Drought in hot weather (maximum supply temperature/minimum supply volume)
Loss of Lake Wylie
Single Active Failure

Procedure OP/0/A/6400/006C (Nuclear Service Water System) Change 224H, adds steps to ensure that Nuclear Service Water flow to Containment Spray Heat Exchangers is isolated and flushing or flow testing of the Nuclear Service Water System to Auxiliary Feedwater System assured makeup is not in service when a Nuclear Service Water Pump is declared inoperable. When a Nuclear Service Water Pump is inoperable, Technical Specification 3.7.8 requires entering an action statement for the D/G made inoperable by the inoperable Nuclear Service Water Pump. With one Nuclear Service Water Pump inoperable and additional flow paths in service beyond those currently assumed in the Nuclear Service Water System one pump flow balance, the operability of the D/Gs may come into question because the resulting Nuclear Service Water flow to the Diesel Jacket Water Cooling heat exchangers may be below that required by the Nuclear Service Water System flow balance. The Nuclear Service Water System flow balance is performed periodically and as a retest to ensure each essential component cooled by the Nuclear Service Water System receives adequate flow in the faulted ESF alignment. The one pump flow balance assumes the following components are in service:

Train related pump house loads
Two Diesel Generator Jacket Water Cooling Heat Exchangers
Two Component Cooling Heat Exchangers
One Containment Spray Heat Exchanger
One Nuclear Service Water to Auxiliary Feedwater Makeup

One Control Room Area Chiller

The Nuclear Service Water System one pump flow balance was verified with flow to the components listed above, plus an additional Nuclear Service Water to Auxiliary Feedwater makeup. The Train A test was successfully performed on 2/11/00, and the Train B test was successfully performed on 2/10/00. The reason for incorporating this additional Auxiliary Feedwater makeup flow is so that it will be assured that the D/Gs will have adequate Nuclear Service Water flow from one Nuclear Service Water Pump even if both Units Auxiliary Feedwater makeup is in service, concurrent with the other loads assumed in the one pump flow balance. Procedure changes were also made to the procedures that could align Nuclear Service Water flow to a Containment Spray Heat Exchanger or Auxiliary Feedwater assured makeup. The changes added the requirement that all four Nuclear Service Water Pumps and all four D/Gs are verified operable before Nuclear Service Water flow could be aligned to the Containment Spray Heat Exchanger or Auxiliary Feedwater assured makeup. The revision to the Nuclear Service Water System procedure ensures that Nuclear Service Water flow to Containment Spray Heat Exchangers is isolated and flushing or flow testing of Nuclear Service Water to Auxiliary Feedwater assured makeup is not in service at the time a Nuclear Service Water Pump is declared inoperable. If the Nuclear Service Water Pump is still inoperable, and an Operator tries to align Nuclear Service Water flow to a Containment Spray Heat Exchanger or Auxiliary Feedwater assured makeup, the changes made to the Containment Spray Heat Exchanger and Auxiliary Feedwater procedures will prevent it. This provides additional assurance that if a Nuclear Service Water Pump is inoperable, the D/Gs will have adequate Nuclear Service Water flow, concurrent with the other loads assumed in the one pump flow balance.

Evaluation: The change to the Nuclear Service Water System procedure, in combination with the Nuclear Service Water flow balance changes and Containment Spray Heat Exchanger and Auxiliary Feedwater assured makeup procedures, ensure the operability of the D/Gs is maintained with one Nuclear Service Water Pump inoperable. Therefore, when a Nuclear Service Water pump is declared inoperable per Technical Spececification 3.7.8, no D/Gs should be declared inoperable assuming that Nuclear Service Water flow to Containment Spray Heat Exchangers is isolated and flushing or flow testing of Nuclear Service Water to Auxiliary Feedwater assured makeup is not in service. If either flow path (Containment Spray or Auxiliary Feedwater) was in service when a Nuclear Service Water Pump became inoperable, the train associated D/Gs were inoperable until the flow path was isolated. Additionally, procedure changes to the Containment Spray and Auxiliary Feedwater procedures ensure neither of these flow paths is placed in service if a Nuclear Service Water Pump is inoperable.

This procedure change ensures the required component flow specified in the Nuclear Service Water System one pump flow balance is maintained, and no additional Nuclear Service Water loads are introduced, Additionally, the Nuclear Service Water system is not evaluated in UFSAR Chapter 15 as an accident initiator, therefore the probability of occurrence of an accident is not increased by this procedure change.

Equipment operation for the Nuclear Service Water, Auxiliary Feedwater, and Containment Spray Systems and Diesel Generators and is unchanged from the previously evaluated revision of this procedure. The procedure change ensures no additional Nuclear Service Water System loads are introduced so that the required component flow and

pressure specified in the Nuclear Service Water one pump flow balance will be maintained.

The Nuclear Service Water, Auxiliary Feedwater, and Containment Spray Systems and Diesel Generators will still be capable of performing their accident mitigation functions. No accidents are identified that will have their radiological consequences affected as a result of this procedure change. The required component flow and pressure specified in the Nuclear Service Water System one pump flow balance will be maintained.

There is no unreviewed safety question associated with the procedure change. No Technical Specification changes are required. No UFSAR changes are required

96 **Type:** Procedure **Unit:** 0

Title: Procedure OP/0/A/6400/006C, Nuclear Service Water System, Change 225C, Modification CE-61633, Nuclear Service Water Pit A Swap Setpoint

Description: Valve 1RN-3A is a part of an assured suction flowpath from the Standby Nuclear Service Water Pond (SNSWP) to the A Train Nuclear Service Water System Pumps. On July 3, 2000 it was determined that the valve was not functioning properly (the valve shows an "intermediate" position indication. The valve is being maintained in the open position with power removed, and the Lake isolation valves 1RN-1A and 1RN-2B are being maintained closed with power removed. This is due to internal problems with valve 1RN-3A. This made it necessary to align the suction and discharge Nuclear Service Water System flowpath to the SNSWP. This is an unacceptable long term solution since the SNSWP will heatup over time. Technical Specifications place an upper limit on the SNSWP temperature of 91.5 degrees F. To reduce the heatup to the SNSWP, the normal Nuclear Service Water System operation is being altered to allow the B Train of Nuclear Service Water to take suction from Lake Wylie and discharge to Lake Wylie. The alignment limitations of the design of the Nuclear Service Water System necessitate that the A Train of Nuclear Service Water System also discharge to Lake Wylie. This means that the suction of A Train of Nuclear Service Water will be aligned to the SNSWP while the discharge is aligned to Lake Wylie. This condition is acceptable as long as A Train of the Nuclear Service Water System is not running. Gravity draining of the SNSWP will not occur since the Pit A Lake isolation valves 1RN-1A and 1RN-2B are maintained closed with power removed. Also, due to the design of the Nuclear Service Water System Pumps, it is impossible for a 26 foot column of water to be maintained in the pump casing and siphon from the pump pit up through the pump discharge isolation valve and check valve, into the Nuclear Service Water System, and ultimately to the lake. The pump discharge isolation valve and check valve are closed when the pump is off. Some events (Ss signal, Loss Of Offsite Power, and Loss of Lake Wylie) automatically start the standby Nuclear Service Water System Pumps. This start of the A Train Nuclear Service Water System Pumps will cause the SNSWP to be pumped to Lake Wylie. To prevent the draining of the SNSWP below Technical Specification limits (571 feet elevation), the A Train Nuclear Service Water System low pit level autoswap will be adjusted (per modification CE-61633) from a current nominal value of 557.5 feet to 571.5 feet. This will ensure an autoswap of the Nuclear Service Water System suction and discharge flowpath from Lake Wylie to the SNSWP prior to reaching the Technical Specification level limits. The time duration from normal SNSWP level to Technical Specification minimum level on the SNSWP is approximately 28.5 hours based on a nominal volume of 42.8 million gallons above the minimum level and a combined flow rate from 1A and 2A Nuclear Service Water System Pump of approximately 25,000 gpm. During this time operator action may stop the A Train Nuclear Service Water System Pumps or the Nuclear Service Water System may be manually aligned to the SNSWP which will stop the level decrease and preclude an automatic actuation of the swapover logic. This alternate Nuclear Service Water System alignment and the supporting modification and procedures will be in place until the next Unit 1 Outage (1EOC12), and will be removed as part of the Nuclear Service Water System Train A work, scheduled for October 19 through October 31, 2000. During this outage, plans are to repair valve 1RN-3A and remove this modification and align the Nuclear Service Water System to its normal configuration.

Evaluation: There is no unreviewed safety question involved with Change 225C to Procedure OP/O/A/6400/006C, Nuclear Service Water System, Minor Modification CE-61633, Nuclear Service Water System Pit A Swap Set Point, and an Operable-But-Degraded evaluation associated with the non-conservative Nuclear Service Water System Pit level transfer set point in Technical Specification 3.3.2, Table 3.3.2-1 Function 10, Nuclear Service Water Suction Transfer - Low Pit Level. No Technical Specification changes are required. No UFSAR Changes are required.

180 **Type:** Procedure **Unit:** 0

Title: Procedure OP/0/A/6400/06F, "Nuclear Service Water System Flush Procedure" Revision 25D

Description: Procedure OP/0/A/6400/06F, "Nuclear Service Water System Flush Procedure" Revision 25D adds steps to allow opening of one Unit 2 Containment Spray System Heat Exchanger on the operable Nuclear Service Water System train to provide an additional flow path for the operating Nuclear Service Water Pump. Unit 1 is shutdown (Mode 5, Mode 6, or No Mode) and Unit 2 is in Mode 1, therefore only Unit 2 can have a design basis LOCA.

Evaluation: Opening a Unit 2 Containment Spray System Heat Exchanger on an operable train of the Nuclear Service Water System is bounded by the one pump Nuclear Service Water System flow balance. The one pump analysis shows that one Nuclear Service Water System Pump has sufficient capacilty to maintain the shutdown unit indefinitely in Cold Shutdown (commencing 36 hours following a trip from rated thermal power) while supplying the post LOCA loads on the other unit. There is no unreviewed safety question associated with this procedure change. No Technical Specification change is required. No UFSAR change is required.

17 **Type:** Procedure **Unit:** 2

Title: Procedure OP/2/A/6100/001 Revision 122 "Controlling Procedure for Unit Startup"

Description: Procedure OP/2/A/6100/001 Rev 122 adds Enclosure 4.12 (Secondary Heatup Checklist). The checklist will be used to isolate various steam system condensate drains, as well as leaking Main Steam Bypass to Condenser System or Main Steam Vent to Atmosphere System dump valves, to support heatup of the plant.

Evaluation: There are three main concerns associated with isolating the steam drains and leaking dump valves during a heatup - water carry over to the turbine, water hammer in the steam lines, and water hammer in the drain lines upon reopening the isolation valve. Revision 122 to OP/2/A/6100/001 provides adequate guidance and precautionary measures to prevent occurrence of these three concerns and to prevent increasing the probability of occurrence of a steam line break (the controlling UFSAR Chapter 15 accident scenario for this activity). Also, the steam dump valves are not required for safe shutdown of the reactor and thus are not nuclear safety related components. Therefore, there are no safety concerns associated with isolating atmospheric or condenser dumps to support unit startup. There are no unresolved safety questions associated with Revision 122 of OP/2/A/61 00/001 as related to isolating steam line drains and leaking steam dump valves to support Unit startup. No Technical Specification changes are required. No UFSAR changes are required.

35 **Type:** Procedure **Unit:** 2

Title: Procedure OP/2/A/6200/001M Revision 13, "Chemical and Volume Control System Fill and Vent"

Description: This revision to Procedure OP/2/A/6200/001M "Chemical and Volume Control System Fill and Vent" Revision 13 addresses keeping valves 2NV-271 and 2NV-291 closed during fill and vent operations. This change to the Unit 2 alternate seal injection flow path valves 2NV-271 and 2NV-291 will help to preclude an undesirable leakage path.

Evaluation: This revision to procedure OP/2/A/6200/001M does not involve an unreviewed safety question. The procedure revision does not affect the probability or consequences of accidents analyzed in the UFSAR. Air entrapment is not a problem due to the piping layout. The alternate seal injection line is not an accident initiator nor is it used to mitigate Design Basis accidents or events. The Reactor Coolant Pump seals are still protected by the normal method of seal injection from the Centrifugal Charging Pump. The normal method of seal injection is assured by safety related valves that can be relied upon to go to their design position in the event of an accident. In the event of a failure of forced seal injection from the Centrifugal Charging Pumps, the Reactor Coolant Pumps have a thermal barrier that would cool the Reactor Coolant System water as it back-flowed through the seals thus providing the same seal cooling as the forced seal injection. Also available in the event of a loss of seal injection and Component Cooling System flow to the Reactor Coolant Pump thermal barrier is the Standby Makeup Pump which can supply seal injection water to the Reactor Coolant Pumps seals from the Spent Fuel Pool. The Standby Makeup Pump has Diesel Generator power to the pump in the event that its normal power supply is not available. No Technical Specifications changes are required. No UFSAR changes are required.

234 **Type:** Procedure **Unit:** 1

Title: Procedure OP/l/A/6200/001M "Chemical and Volume Control System Fill and Vent" Revision 10

Description: Procedure OP/l/A/6200/001M "Chemical and Volume Control System Fill and Vent" was revised to optimize the fill and vent operations associated with returning the Chemical and Volume Control System to service following maintenance. The fill and vent of the Chemical and Volume Control System is performed in "No Mode". Since the fuel is in the Spent Fuel Pool, the Spent Fuel Pool Cooling System, the Component Cooling System, and the Nuclear Service Water Systems are protected since they provide the cooling to the core. Thus, the fill and vent of the Chemical and Volume Control System will not affect the systems that are in service to cool the core. There are no direct references to fill and vent of the Chemical and Volume Control System in the UFSAR.

The water supply source for this evolution is from the Refueling Water Storage Tank. This procedure revision reroutes the water from this tank directly to the Chemical and Volume Control System Pump suction header. In the past, the Refueling Water Storage Tank water was routed to the Chemical and Volume Control System through Residual Heat Removal System Train 1A. Residual Heat Removal System Train 1A could be aligned to the Reactor Coolant System Loops if enough water was available. However the procedure also listed the preferred alignment as coming from the Refueling Water Storage Tank. The Refueling Water Storage Tank has normally been the fill source in past outages. Filling and venting the Chemical and Volume Control System directly from the Refueling Water Storage Tank instead of through the Residual Heat Removal System can increase the fill rate since the water is fed directly to the Chemical and Volume Control System pump suction. The fill and vent methodology is the same as in the past for the Chemical and Volume Control System pump suction, the pumps, normal charging line, and the piping up to valves NI-9A and NI-10B.

This procedure change also combines the old enclosure 4.7 (Fill and Vent of Chemical and Volume Control System Piping Between valve 1NV-831 and Reactor Coolant System Loop/Pressurizer Isolations) and old enclosure 4.9 (Fill and Vent of Chemical and Volume Control System Piping Between and Reactor Coolant System Loop/Pressurizer Isolations and Reactor Coolant System) into a new enclosure 4.7 (Fill and Vent of Chemical and Volume Control System Piping Between valve 1NV-831 and the Reactor Coolant System). This streamlines the process by simply extending the boundaries to the Reactor Coolant System instead of having an intermediate step at the Reactor Coolant System /Pressurizer Loop Isolations. This change also necessitated combining the old valve checklists of enclosure 4.8 and 4.10 into a new valve checklist that is kept in enclosure 4.8.

Evaluation: There is no unreviewed safety question associated with this procedure revision. Since this evolution is performed in "No Mode", there will be no effect on the probability or consequences of accidents evaluated in the UFSAR. No Technical Specification revisions are required. No UFSAR changes are required.

59 **Type:** Procedure **Unit:** 0

Title: Procedure PT/0/A/4400/008A, Nuclear Service Water System Flow Balance Train A, Revision 34 Change B

Description: The Nuclear Service Water System, including Lake Wylie and the Standby Nuclear Service Water Pond, is the ultimate heat sink for various nuclear safety related heat loads during normal operation and design basis events. The Nuclear Service Water System supports Emergency Core Heat Removal operation by providing cooling to the Component Cooling System via the Component Cooling Heat Exchangers and also to the Diesel Generators via the Diesel Generator Engine Jacket Water Cooler System Heat Exchangers. Other nuclear safety related loads include the Containment Spray Heat Exchanger and Control Room Chiller Condenser. The Nuclear Service Water System also provides assured makeup to the Component Cooling System, Spent Fuel Pool, Auxiliary Feedwater supply and the Containment Seal Water Injection System.

The Nuclear Service Water System shall be capable of mitigating the consequences of a design basis event on one unit concurrent with a loss of offsite power on both units, assuming a single failure. The most stringent design basis event for the Nuclear Service Water System is a Loss of Coolant Accident (LOCA) on one unit. The following assumptions are further postulated for the Nuclear Service Water System design basis LOCA:

1. Normal Shutdown of the remaining unit from normal operation
2. Loss of Non-Emergency A/C Power (loss of offsite power) affecting both units
3. Prolonged Drought in hot weather (maximum supply temperature/ minimum supply volume)
4. Loss of Lake Wylie
5. Single Active Failure

Revision 34 change B to the Nuclear Service Water System Flow Balance Train A procedure, PT/0/A/4400/08A, will add an additional 500 gpm Nuclear Service Water System to Auxiliary Feedwater system makeup flow to the component flows assumed in the Nuclear Service Water System One Pump Flow Balance. The Nuclear Service Water System flow balance is performed periodically and as a retest to ensure each essential component cooled by Nuclear Service Water System receives adequate flow in the faulted ESF alignment. The one pump flow balance assumes the following components are in service:

Train related pump house loads
Two Diesel Generator Jacket Water Cooler Heat Exchangers
Two Component Cooling Heat Exchangers
One Containment Spray Heat Exchanger
One Nuclear Service Water System to Auxiliary Feedwater System makeup
One Control Room Chilled Water System Chiller

With one Nuclear Service Water System pump inoperable and additional flow paths in service beyond those currently assumed in the Nuclear Service Water System one pump

balance, the operability of the D/Gs may come into question because the resulting Nuclear Service Water System flow to the Diesel Generator Jacket Water Cooling System heat exchangers may be below that required by the Nuclear Service Water System flow balance. When a Nuclear Service Water pump is inoperable, Tech Specs require entering an action statement for the D/G made inoperable by Nuclear Service Water System. Trended flow balance data indicates sufficient margin exists in component flow and required header pressure to model Nuclear Service Water System to Auxiliary Feedwater System makeup for both units. The advantage to incorporating this additional flow is that it will be assured that the D/Gs will have adequate Nuclear Service Water System flow from one Nuclear Service Water System pump even if both Units Auxiliary Feedwater System makeup is in service, concurrent with the other loads assumed in the one pump flow balance.

Evaluation: Changing the Nuclear Service Water System flow balance to incorporate an additional Auxiliary Feedwater System makeup does not change the method of operation or the design basis of the Nuclear Service Water System system. The required Nuclear Service Water System flow and pressure to the components in the flow balance will be maintained with the additional Auxiliary Feedwater System demand of 500 gpm. The assumed Nuclear Service Water System to Auxiliary Feedwater System makeup flow rate of 500 gpm comes from CNC-1223.24-00-0011, Nuclear Service Water System Test Acceptance Criteria, Rev 20.

The additional 500 gpm will be made up from a combination of two sources: (1) the flow margin from other components and (2) from an increase in flow from the Nuclear Service Water System pump. Trended flow balance data indicates sufficient margin exists in component flow and required header pressure to add another 500 gpm, assuming the components supplied all of the additional flow. The flow balance data also supports another 500 gpm from a Nuclear Service Water System pump, assuming the pump supplied all of the 500 gpm additional flow. The total flow from a Nuclear Service Water System pump during the flow balance is typically between 18,000 and 20,000 gpm. An additional 500 gpm would not cause the pump to approach run out flows of approximately 25,000 gpm.

There are no unreviewed safety questions associated with this Procedure change. No Technical Specification changes are required. No change is required to UFSAR..

60 **Type:** Procedure **Unit:** 0

Title: Procedure PT/0/A/4400/008B, Nuclear Service Water System Flow Balance Train B, Revision 30 Change D

Description: The Nuclear Service Water System, including Lake Wylie and the Standby Nuclear Service Water Pond, is the ultimate heat sink for various nuclear safety related heat loads during normal operation and design basis events. The Nuclear Service Water System supports Emergency Core Heat Removal operation by providing cooling to the Component Cooling System via the Component Cooling Heat Exchangers and also to the Diesel Generators via the Diesel Generator Engine Jacket Water Cooler System Heat Exchangers. Other nuclear safety related loads include the Containment Spray Heat Exchanger and Control Room Chiller Condenser. The Nuclear Service Water System also provides assured makeup to the Component Cooling System, Spent Fuel Pool, Auxiliary Feedwater Supply and the Containment Seal Water Injection System.

The Nuclear Service Water System shall be capable of mitigating the consequences of a design basis event on one unit concurrent with a loss of offsite power on both units, assuming a single failure. The most stringent design basis event for the Nuclear Service Water System is a Loss of Coolant Accident (LOCA) on one unit. The following assumptions are further postulated for the Nuclear Service Water System design basis LOCA:

1. Normal Shutdown of the remaining unit from normal operation
2. Loss of Non-Emergency A/C Power (loss of offsite power) affecting both units
3. Prolonged drought in hot weather (maximum supply temperature/ minimum supply volume)
4. Loss of Lake Wylie
5. Single Active Failure

Revision 30 Change D to the Nuclear Service Water System Flow Balance Train B procedure, PT/0/A/4400/08B, will add an additional 500 gpm Nuclear Service Water System to Auxiliary Feedwater System makeup flow to the component flows assumed in the Nuclear Service Water System One Pump Flow Balance. The Nuclear Service Water System flow balance is performed periodically and as a retest to ensure each essential component cooled by Nuclear Service Water System receives adequate flow in the faulted ESF alignment. The one pump flow balance assumes the following components are in service:

Train related pump house loads
Two Diesel Generator Jacket Water Cooler Heat Exchangers
Two Component Cooling Heat Exchangers
One Containment Spray Heat Exchanger
One Nuclear Service Water System to Auxiliary Feedwater System Makeup
One Control Room Chilled Water System Chiller

With one Nuclear Service Water System pump inoperable and additional flow paths in service beyond those currently assumed in the Nuclear Service Water System one pump

balance, the operability of the Diesel Generators may come into question because the resulting Nuclear Service Water System flow to the Diesel Generator Jacket Water Cooling System heat exchangers may be below that required by the Nuclear Service Water System flow balance. When a Nuclear Service Water Pump is inoperable, Technical Specifications require entering an action statement for the Diesel Generator made inoperable by Nuclear Service Water System. Trended flow balance data indicates sufficient margin exists in component flow and required header pressure to model Nuclear Service Water System to Auxiliary Feedwater System makeup for both units. The advantage to incorporating this additional flow is that it will be assured that the Diesel Generators will have adequate Nuclear Service Water System flow from one Nuclear Service Water System Pump even if both Units Auxiliary Feedwater System makeup is in service, concurrent with the other loads assumed in the one pump flow balance.

Evaluation: Changing the Nuclear Service Water System flow balance to incorporate an additional Auxiliary Feedwater System makeup does not change the method of operation or the design basis of the Nuclear Service Water System System. The required Nuclear Service Water System flow and pressure to the components in the flow balance will be maintained with the additional Auxiliary Feedwater System demand of 500 gpm. The assumed Nuclear Service Water System to Auxiliary Feedwater System makeup flow rate of 500 gpm comes from CNC-1223.24-00-001 1, Nuclear Service Water System Test Acceptance Criteria, Rev 20.

The additional 500 gpm will be made up from a combination of two sources: (1) the flow margin from other components and (2) from an increase in flow from the Nuclear Service Water System Pump. Trended flow balance data indicates sufficient margin exists in component flow and required header pressure to add another 500 gpm, assuming the components supplied all of the additional flow. The flow balance data also supports another 500 gpm from a Nuclear Service Water System Pump, assuming the pump supplied all of the 500 gpm additional flow. The total flow from a Nuclear Service Water System Pump during the flow balance is typically between 18,000 and 20,000 gpm. An additional 500 gpm would not cause the pump to approach run out flows of approximately 25,000 gpm.

There are no unreviewed safety questions associated with this Procedure change. No Technical Specification changes are required. No change is required to UFSAR.

213 **Type:** Procedure **Unit:** 0

Title: Procedure PT/0/A/4600/102, Revision 0, Concrete Compressive Strength Conformation

Description: Procedure PT/0/A/4600/102 directs the use of a small hand held device, a spring loaded concrete rebound hammer, to estimate the compressive strength of concrete in an existing structure. The plunger of the device impacts concrete and rebounds a measured distance. The distance of rebound provides a correlation to the concrete compressive strength. The concrete is not damaged by the action of the rebound hammer. The activity itself has no effect on the plant structure being examined or on any mechanical or electrical systems. There are no system isolations required by this procedure. A prerequisite for the procedure is to have the coating (paint) removed from an area of concrete of about two square feet. This does not have any effect on concrete compressive strength or on design considerations for the structure involved.

Evaluation: There are no unreviewed safety questions associated with this procedure. There are no plant system alignments or changes to methods of operation that are required or affected by this procedure. There is no potential for damage to systems, structures or components. There are no accident scenarios or accident analyses that could be affected by this procedure. There are no potential system interactions created by the procedure. There are no alterations to seismic or environmental qualifications caused by the procedure. There are no alterations to qualification for any other natural phenomena caused by the procedure. This procedure has no effect on any structure, system or component. There are no accidents evaluated in the UFSAR that are affected by the procedure. No Technical Specification changes are required. No UFSAR changes are required

244 **Type:** Procedure **Unit:** 1

Title: Procedure PT/1/A/4150/001 D, "Reactor Coolant System Leakage Calculation", Revision 42

Description: Procedure PT/1/A/4150/001 D, "Reactor Coolant System Leakage Calculation", Revision 42 will change the method for performing this calculation.

The Unit 1 Pressurizer Relief Tank (PRT) Level instrumentation is providing erratic indication, affecting the accuracy of the identified leakage portion of the Reactor Coolant System Leakage Calculation. During a normal sixty minute run of the Operator Aid Computer (OAC) Reactor Coolant System leakage program, level in the PRT has increased or decreased as much as 0.05%. This level change is affecting the calculation results by +/- 0. 11 gpm, and is contributing to the frequency at which the calculation is producing negative unidentified leakage results. Over longer periods of time, the Unit 1 PRT level is fluctuating, but not establishing the steady increase suggested by the leakage calculation. Repetitive runs of the OAC leakage program have confirmed that the program is being affected by the fluctuations in PRT level instrumentation, without a change in actual PRT level. A work request has been issued to troubleshoot and repair the Unit 1 PRT level instrumentation; however, the level transmitter is located in lower containment where access is limited.

To compensate for the indication fluctuations, Procedure PT/l/A/4150/001 D, Reactor Coolant System Leakage Calculation, will be revised such that leakage into the Unit 1 PRT is not considered Identified Leakage. This will be accomplished by inserting a fixed level indication into the leakage program. The OAC leakage program will include leakage into the Unit 1 PRT with Unidentified Leakage. This is conservative reporting with any actual leakage into the PRT resulting in higher Unidentified Leakage. Technical Specification 3.4.13 limits Unidentified Leakage at less than 1.0 gpm.

As a result of this procedure change, the OAC leakage program will report Identified Leakage at less than the actual value. However, there will be no change in the way the OAC program reports Reactor Coolant System Total Leakage.

Evaluation: The purpose for monitoring Reactor Coolant System operational leakage is to protect the reactor coolant pressure boundary from degradation and the core from inadequate cooling. Technical Specification 3.4.13, "Reactor Coolant System Operational Leakage", requires that Reactor Coolant System leakage be within limits of 1 gpm unidentified and 10 gpm identified. 10CFR50, Appendix A, GDC 30, requires means for detecting and, to the extent practical, identifying the source of reactor coolant leakage.

The safety significance of Reactor Coolant System leakage varies widely depending on its source, rate, and duration. Quickly separating the identified leakage from the unidentified leakage is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur. By ensuring proper leak detection and identification, appropriate measures are established to protect the reactor coolant system pressure boundary from degradation and the core from inadequate cooling, which ensures barrier protection preventing the accident analysis radiation release assumptions from being exceeded. Reporting leakage into the PRT as Unidentified Leakage places tighter limits on system leakage, which ensures the appropriate operator attention is provided.

Per the basis of Technical Specification 3.4.13, the consequences for exceeding the operational leakage limits include the possibility of a Loss of Coolant Accident (LOCA). This procedure change does not affect the calculation of total leakage. The change reduces the uncertainty of identification of leakage into the PRT due to the instability of the PRT level indication. By removing the uncertainty in PRT level changes, the calculation is more conservative assuring that operator action will be taken to correct leakage problems. Any actual leakage into the PRT will be conservatively accounted for as Unidentified Leakage. Therefore, the probability of a LOCA due to ineffective leakage monitoring is not increased.

No accident evaluated in Chapter 15 of the UFSAR would be affected by this procedure change. This procedure change will conservatively report leakage into the Unit 1 PRT as Unidentified Leakage. Both identified and unidentified leakage will be maintained within the allowable limits specified in Technical Specification 3.4.13, Reactor Coolant System Operational Leakage.

This procedure will be used to quantify leakage from the reactor coolant system as required by Technical Specification 3.4.13. This procedure change will conservatively report leakage into the Unit 1 PRT as Unidentified Leakage, and maintain reactor coolant system leakage within the allowable limits specified in Technical Specification 3.4.13. Therefore, this activity will not create the possibility of an accident of a different type than any evaluated in the UFSAR.

This procedure change will allow inserting a fixed PRT level into the OAC leakage program. This change is made within the OAC, and does not physically change equipment or plant operation. By conservatively reporting leakage into the Unit 1 PRT as Unidentified Leakage, the control room response to any change in leakage will also be conservative. Therefore, this activity does not increase the probability of a malfunction of equipment important to safety evaluated in the UFSAR.

This procedure change will result in the reporting of leakage into the PRT as Unidentified Leakage. As such, the control room response to any change in leakage will be conservative. Therefore, this activity does not increase the consequences of a malfunction of equipment important to safety evaluated in the UFSAR.

There are no unreviewed safety questions associated with this procedure revision. No Technical Specification changes are required. No UFSAR changes are required.

235 Type: Procedure **Unit:** 0

Title: Procedure PT/1/A/4200/018 "Valve 1NV-849 Travel Stop Verification" Revision 0 and PT/2/A/4200/018 "Valve 2NV-849 Travel Stop Verification" Revision 1

Description: Procedure PT/1/A/4200/018 (Valve 1NV-849 Travel Stop Verification) was developed from the revised procedure developed for Unit 2 to ensure that the travel stops to limit the maximum allowable flow are set properly following maintenance on valve 1NV-849 (Normal Letdown Variable Orifice). This procedure will be run when the Reactor Coolant System is at normal operating temperature and pressure. Valve 1NV-849 will then be placed in service by itself. Normal letdown flow will then be increased until valve 1NV-849 is either wide open or a flow of 110 gallons per minute (gpm) is reached, whichever comes first. Maintenance will then adjust the travel stops on the valve such that only 110 gpm is the maximum allowable flow. Normal letdown flow is then reduced as desired. The procedure will then allow Operations to restore the desired alignment of the letdown orifices.

Procedure PT/2/A/4200/081 (Valve 2NV-849 Travel Stop Verification) was changed to incorporate lessons learned from running the procedure during the 2EOC10 Refueling Outage. The procedure was changed to reduce letdown flow through valve 2NV-849 and then return it to 110 gpm in order to verify the correct setting on the limit switches for valve 2NV-849.

Work orders were developed and incorporated into the new Unit 1 procedure and into the Unit 2 procedure revision. The work orders will be used to cover the maintenance work that may be required if the travel stops and/or limit switches have to be adjusted.

Evaluation: There is no unreviewed safety question associated with this procedure revision. The unconditioned use of valve NV-849 is not an accident initiator as described in the UFSAR. This valve was intended for unconditioned use in the original design of the plant. This procedure supports returning this valve to its original purpose. Valve NV-849 is not a nuclear safety related valve. Its flowpath is isolated by nuclear safety related components in the event of an accident. No Technical Specification revisions are required. No UFSAR changes are required.

190 **Type:** Procedure **Unit:** 1

Title: Procedure PT/1/A/4200/01N "Reactor Coolant System Pressure Boundary Leak Rate Test", Revision 43

Description: This revision to the Reactor Coolant Pressure Boundary Valve Leak Rate Test procedure involves (1) optimization of the test sequence, (2) adding an alternative for the Residual Heat Removal System Suction Isolation Valve Test in Mode 4, and (3) separating the Valve Checklist into two enclosures (i.e., separating remote and manual operated valves). This revision also enhances the readability of the procedure by converting to the most recent guidelines provided in the Catawba Standard Template and Catawba Procedures Writers Manual. During this retype additional format changes were made to enhance the procedures. Some of these enhancements were administrative in nature, others were to provide clarification to the procedure user.

Evaluation: None of these changes affected the test method or acceptance criteria of the test. Effect on plant operation systems and design limits have been adequately considered. No unreviewed safety questions are created as a result of this procedure revision. No Technical Specification changes are required. No UFSAR changes are required.

36 **Type:** Procedure **Unit:** 0

Title: Procedure PT/1/A/4200/09 Revision 166P, Procedure PT/2/A/4200/09 Revision 141C and UFSAR Table 7-15

Description: Procedure PT/1/A/4200/09 Revision 166P, Procedure PT/2/A/4200/09 Revision 141C and UFSAR Table 7-15 are being revised. Problems were identified regarding verification of ESF Response Times for the Containment Air Return and Hydrogen Skimmer System as well as errors associated with the ESF Response Times for the Hydrogen Skimmer System suction valves. The Hydrogen Skimmer System suction valve response times were determined to be incorrect because credit for the diesel generator startup and load sequencer delays were not being taken in the Containment Air Return and Hydrogen Skimmer System performance tests, PT/1(2)/A/4450/005A(B). The Containment Air Return and Hydrogen Skimmer System performance tests were subsequently revised to take credit for the diesel generator startup and load sequencer delays.

The Hydrogen Skimmer System suction valves ESF Response Time are now being changed to indicate that the response time terminates when the valves start to open.

The Hydrogen Skimmer System suction valve response times will be changed in the ESFAS Periodic Test Procedures PT/1(2)/A/4200/09, UFSAR Table 7-15 (Note 9), and the Containment Air Return and Hydrogen Skimmer System Design Basis Specification (CNS-1557.VX-00-0001). These changes will be made to clarify that these valves are only required to start opening within 600 seconds after a Sp signal. Procedures PT/1(2)/A/4200/09 will also be revised to clarify that the ESF Response Times for the Containment Air Return and Hydrogen Skimmer Fan operation are verified using a combination of procedures.

Evaluation: There is no unreviewed safety question associated with revising these documents to provide correct information. These changes will have no effect on the probability or consequences of accidents evaluated in the UFSAR. No Technical Specification changes are required. UFSAR Table 7-15 will be revised.

192 **Type:** Procedure **Unit:** 1

Title: Procedure PT/1/A/4200/13H, "Safety Injection System and Chemical and Volume Control System Check Valve Test", Revision 25

Description: Revision 25 of Procedure PT/1/A/4200/13H, "Safety Injection System and Chemical and Volume Control System Check Valve Test", changes the procedure format to fit the current computer document management system template. The "Enclosure" format has been dropped and all steps (and supporting Enclosures) are sequentially numbered. This change results in procedure steps for section 12 being performed as stand alone steps and not as part of Enclosure 13.1. Changes have been made to the Reference section (Section 2.0) to include the acceptance criteria verification calculations that document the values in Section 11.0. The procedure time requirements have been updated to reflect the larger amount of testing performed under this procedure. Finally certain steps for static/DP testing have been relocated to make the procedure more efficient. These changes are editorial and do not affect the intent or the actual performance of the procedure. This evaluation will not cover these editorial changes.

Non-editorial changes covered under this evaluation

The procedure has been revised throughout to remove all references to the DLOG data acquisition system. The new system (LABVIEW) information and requirements have been inserted. Prerequisite steps have been added to verify the LABVIEW executable files.

Additional pre-requisite (Section 8) steps revised or added are as follows:

The temperature range allowed for the test was reduced to reflect the range analyzed in the equipment measurement uncertainty calculations

Acceptance Criteria (Section 11) changes were made to reflect the values listed in the newly revised QA-1 calculation for acceptance criteria verification.

Procedure body (Section 12) changes are as follows:

Static stroke requirements were added for valves that got DP strokes later in the procedure

Steps were added to ensure simulated seal injection flow of 35 +/- 5 gpm when performing Chemical and Volume Control System flow verifications to be consistent with ECCS flow balance requirements.

A step that involved closing valve 1NI-136B was re-sequenced so that the Safety Injection Pump miniflow valves could be opened due to an interlock with this valve. This was a lesson learned from the last Unit 1 outage.

The Chemical and Volume Control System injection header flow instrumentation requirement was changed to allow the use of the control room gauge if desired to meet the acceptance criteria verification of valve 1NV-254.

Steps were added to compare Safety Injection Pump cold leg flow with the hot leg isolation valve bonnet vent valves in the open and closed position. This test ensures the modified hot leg isolation valves do not leak by at a rate that would adversely impact the cold leg flow balance. This change results from a modification that installed manual bonnet vents on valve 1NI-121A and 1NI-152B to eliminate pressure locking/thermal binding concerns.

Evaluation: The purpose of procedure PT/1/A/4200/13H, "Safety Injection System and Chemical and Volume Control System Check Valve Test", is to comply with Catawba IST program requirements for operability (full & partial stroke exercise) for those valves listed in the procedure. The reactor vessel is open with no fuel in the core during performance of this test. Safety Injection System and Chemical and Volune Control System pumps discharge into the reactor vessel and water is allowed to overflow into the reactor vessel cavity. Since the testing is performed while the Unit is in No Mode, none of the ECCS systems are required to be operable during this time. Limits and precautions are in place to ensure that the pumps are protected from runout. Flow below the runout limit is assured by throttle valve position that will not have changed since the last full ECCS flow balance.

Data acquisition system changes do not adversely affect the intended purpose of this test. The new data acquisition system has been approved for nuclear safety related applications under the Duke Power SDQA program. The changes made in the data acquisition program necessitated an update to the calculations that document the procedure's acceptance criteria. These nuclear safety related calculation changes have been approved and the Acceptance Criteria section updated appropriately. Adjusting the acceptance criteria ensures that all valve strokes are completed successfully given the measurement uncertainty of the test equipment. The same calculation that verifies the procedure acceptance criteria also allows the use of control room indication for Chemical and Volume Control System injection flow when stroking valve 1NV-254. The error assumed for Chemical and Volume Control System injection flow rate in the acceptance criteria verification is consistent with the expected error associated with the control room gauge. Using test instrumentation for this application is also acceptable since the resulting instrument error would be smaller and the acceptance criteria still bounding.

Re-sequencing steps so that valve interlocks can be met or adding steps for valve static strokes do not adversely affect any equipment required operable during No Mode. None of the ECCS equipment operated as part of this test is required to be operable in No Mode, and care has been taken to avoid pump deadheading or runout. Even though ECCS equipment is not required to be operable during the test, the changes described above will not compromise any ECCS components while they are in operation.

Re-sequencing and adding steps to record flow with the hot leg isolation valve bonnet vents open and closed assures that unacceptable leakage across the hot leg isolation valves does not occur due the implementation of modification CN-11385. This modification added bonnet vents to hot leg isolation valves NI-121A and NI-152B. This change will not compromise the safe operation of any ECCS components and assures cold leg injection flow is consistent with the requirements of the Safety Analysis.

Finally, simulating seal injection flow through the normal charging line will not adversely affect the Chemical and Volume Control System Pump(s) in operation. Throttle valve position in each Chemical and Volume Control System cold leg injection line prevents

runout of a single pump because the original balance was performed with simulated seal injection flow. The procedure steps were added to be consistent with the ECCS flow balance procedure. Likewise, putting flow down the charging line will not affect any components in this pathway. A simulated seal injection flow rate of 35 +/- 5 gpm is much less than the maximum allowable flow rate through the regenerative heat exchanger (150 gpm).

This procedure revision involves no unreviewed safety questions. No Technical Specification changes are required. No UFSAR changes are required.

225 **Type:** Procedure **Unit:** 1

Title: Procedure PT/1/A/4250/003E "Auxiliary Feedwater System Discharge Control Valve Throttling Procedure", Revision 24A

Description: Procedure PT/1/A/4250/003E "Auxiliary Feedwater System Discharge Control Valve Throttling Procedure" is being changed to allow the use of error adjusted acceptance criteria.

Evaluation: There are no accidents evaluated in the UFSAR which assume that the Auxiliary Feedwater System contributes as an accident initiator. Feedwater malfunctions that increase Main Feedwater flow are evaluated, but no inadvertent actuation of the Auxiliary Feedwater System. The method of performance of the procedure is not changed. Therefore, this procedure change will not increase the probability or consequences of accidents evaluated in the UFSAR. There are no unreviewed safety questions associated with this procedure revision. No Technical Specification changes are required. No UFSAR changes are required.

191 **Type:** Procedure **Unit:** 0

Title: Procedure PT/1/A/4400/009, "Cooling Water Flow Monitoring for Asiatic Clams and Mussels Quarterly Test (Unit 1)", Revision 48A and PT/2/A/4400/009, "Cooling Water Flow Monitoring for Asiatic Clams and Mussels Quarterly Test (Unit 2)", Revision 28A

Description: Restricted change (Revision 48A) is being made to Procedure PT/1/A/4400/009, "Cooling Water Flow Monitoring for Asiatic Clams and Mussels Quarterly Test (Unit 1)", and restricted change (Revision 28A) is being made to PT/2/A/4400/009, "Cooling Water Flow Monitoring for Asiatic Clams and Mussels Quarterly Test (Unit 2)". These restricted changes only apply when:

Unit 1 is shut down (Mode 5, 6 or No-Mode)
Unit 2 is at power
Heat Exchanger D/P testing is only performed on the inoperable Nuclear Service Water System Train
Nuclear Service Water System Trains are separated

Evaluation: The Nuclear Service Water System including Lake Wylie and the Standby Nuclear Service Water Pond (SNSWP), is the ultimate heat sink for various nuclear safety related heat loads during normal operation and design basis events. The Nuclear Service Water System supports Emergency Core Heat Removal (ECCS) operation by providing cooling to the Component Cooling System via the Component Cooling heat exchangers and also to the Diesel Generators (D/Gs) via the D/G Engine Jacket Water Cooler system heat exchangers. Other nuclear safety related loads include the Containment Spray Heat Exchanger and Control Room Chilled Water System Chiller Condenser. The Nuclear Service Water System also provides assured makeup to the Component Cooling System, Spent Fuel Pool, Auxiliary Feedwater supply, and the Containment Seal Water Injection system.

The Nuclear Service Water System shall be capable of mitigating the consequences of a design basis event on one unit concurrent with a Loss of Non-Emergency A/C Power (loss of offsite power) affecting both units, assuming a single active failure. The most stringent design basis event for the Nuclear Service Water System is a Loss of Coolant Accident (LOCA) on one unit. The following assumptions are further postulated for the Nuclear Service Water System design basis LOCA:

Normal shutdown of the remaining unit from normal operation
Prolonged drought in hot weather (maximum supply temperature/minimum supply volume)
Loss of Lake Wylie

Restricted change (Rev 48A) to PT/1A/4400/009, "Cooling Water Flow Monitoring for Asiatic Clams and Mussels Quarterly Test (Unit 1)", and Restricted change (Rev 28A) to PT/2/A/4400/009, "Cooling Water Flow Monitoring for Asiatic Clams and Mussels Quarterly Test (Unit 2)" delete the prerequisite conditions to require four Operable Nuclear Service Water System pumps and four Operable D/Gs, and adds steps to ensure that Unit 1 is shut down (Mode 5, Mode 6 or No-Mode), Unit 2 is at power, Heat Exchanger D/P testing is only performed on the inoperable Nuclear Service Water System train, and Nuclear Service Water System trains are separated. The entire train of the

Nuclear Service Water System associated with the train being tested is considered inoperable. The Operable train is unaffected by the Heat Exchanger D/P tests on the inoperable train, since the trains will be separated. The Operable Nuclear Service Water System train is bounded by the Nuclear Service Water System one pump flow balance. The requirement for four Operable Nuclear Service Water System pumps and D/Gs is not applicable to the inoperable train, therefore these requirements can be deleted when:

Unit 1 is shut down (Mode 5, Mode 6 or No-Mode)
Unit 2 is at power
Heat Exchanger D/P testing is only performed on the inoperable Nuclear Service Water System Train
Nuclear Service Water System trains are separated.

No unreviewed safety questions are created as a result of this procedure revision. No Technical Specification changes are required. No UFSAR changes are required.

179 **Type:** Procedure **Unit:** 1

Title: Procedure PT/1/A/4400/01, "ECCS Flow Balance" Revision 37

Description: Procedure PT/1/A/4400/01, "ECCS Flow Balance" Revision 37 makes changes to the existing acceptance criteria for each section. These changes are required due to a recent change in the error analysis calculation for flow balance testing and pump head curve verification. In addition steps have been added to the Residual Heat Removal Pump performance sections (Enclosure 13.1) to allow running inservice pumps tests in conjunction with the residual heat removal system flow verification. Also each test enclosure has been revised to reflect the use of a new data acquisition tool. Procedure test data is no longer gathered using the Dlog computer program. An application called "LABVIEW" is used instead. Other editorial changes have been made. These editorial changes do not significantly affect the test procedure or the data and results obtained from the test.

Evaluation: This procedure change has no adverse effect on plant safety. The test is performed with no fuel in the Reactor Core and it does not affect equipment associated with the Spent Fuel Pool. The original purpose of the test is still applicable and the intent of the test is not compromised in any way as a result of these changes. There is no unreviewed safety question associated with the change. No Technical Specification changes are required . No UFSAR changes are required.

246 **Type:** Procedure **Unit:** 1

Title: Procedure PT/1/A/4400/013 "Component Cooling System Pump Interaction Test"

Description: Procedure PT/1/A/4400/013 "Component Cooling System Pump Interaction Test" is being deleted. The procedure was used in testing the miniflow characteristics of the Component Cooling Pumps.

The Catawba Component Cooling System acts as a safety related closed loop treated water system to dissipate waste heat from motor coolers and intersystem heat exchangers serving various systems. In addition, the Component Cooling System provides cooling to engineered safeguards loads after a Design Basis Event. This system serves as a boundary between the Reactor Coolant System and the Nuclear Service Water System; reducing the probability of radioactivity leakage into the environment.

NRC IE Bulletin 88-04 requested that all licensees investigate and correct two miniflow design concerns. The first involves the potential for dead-heading one or more pumps in safety-related systems that have a miniflow line common to two or more pumps of other configurations that do not preclude pump-to-pump interaction during miniflow operation. The second issue is whether or not the installed miniflow capacity is adequate for even a single pump in operation.

Included in the Catawba response to Bulletin 88-04 was a commitment to perform detailed calculations to identify pump flows resulting from appropriate valve alignments while imposing IWP acceptance criteria, to investigate short term minimum flow requirements from the manufacturer, and to evaluate possible setpoint changes to preclude unacceptable pump interaction.

As a result of these commitments, the minimum flow setpoints were changed and the procedure PT/1/A/4400/013 "Component Cooling System Pump Interaction Test" was developed to verify the new setpoints. This procedure failed to produce adequate results and was not able to verify that no interaction existed. Instead of performing this test, analyses have been performed in association with test procedure PT/1/A/4400/03F, "Head Curve Test for Component Cooling System Pumps 1A1, 1A2, 1B1, and 1B2" to ensure a strong pump/weak pump interaction does not exist. The pump head curve tests and Interaction Analysis are adequate to meet the Catawba IE Bulletin 88-04 commitment.

Evaluation: There are no unreviewed safety questions associated with deletion of this procedure. The probability of an accident previously evaluated in the UFSAR will not be increased. The Component Cooling System is a safety related system used to mitigate design basis events. It is not an accident initiator and therefore, deleting the procedure will not increase the probability of an accident previously evaluated in the UFSAR. Deleting the test procedure is an administrative change that does not affect the operation of the system.

The probability of a malfunction of equipment important to safety will not be increased. The procedure being deleted was ineffective in determining whether a strong pump/weak pump interaction exists. Deleting the procedure will therefore not increase the probability of a malfunction. Strong pump/weak pump interaction has been analytically evaluated in several engineering calculations. Where it was found that interaction potential exists, administrative controls have been put in place to prevent the alignments from occurring.

In general, the safety related Component Cooling Pump minimum flow line and control valve provides adequate protection for all credible Component Cooling System alignments during normal operation and design basis response to accidents. Administrative Controls are in place to avoid system alignments where a strong pump/weak pump interaction may occur in excess of that protection provided by the minimum flow path. No Technical Specification changes are required. No UFSAR changes are required.

19 **Type:** Procedure **Unit:** 0

Title: Procedure PT/1/A/4450/03C Revision 39 and PT/2/A/4450/03C Revision 24

Description: Procedures PT/l/A/4450/003C Revision 39 and PT/2/A/4450/003C Revision 24 are being revised. The major changes incorporated into the subject procedures include the following items:

1) In the section about "Prerequisite System Conditions", the step to ensure a calculation has been performed which bounds the radiological dose consequences associated with the vacuum decay test in Modes 1 - 4 is being deleted for Enclosures 13.9 through 13.17.

2) In the section about "Prerequisite System Conditions", add the following step for Enclosures 13.9 through 13.17:

"If Unit 1(2) is in Modes 1 - 4, inform the Unit WCC (Work Control Center) SRO that the Reactor Building is inoperable (LCO 3.6.16)."

In the section about "Prerequisite System Conditions", change the nomenclature for the steps associated with declaring the Annulus Ventilation System train inoperable to match (i.e, Unit/WCC SRO) for Enclosures 13.9 through 13.17.

3) In the section about "Prerequisite System Conditions", the requirement to implement the Compensatory Action for Annulus Ventilation System Performance Test in Modes 1 - 4 after pressure boundary maintenance has been performed is being deleted for Enclosures 13.9 through 13.17. The step in each individual vacuum decay enclosure to establish phone communications between the Control Room Operator and test personnel is also being deleted. These steps are associated with the subject compensatory action.

4) In the section about "Prerequisite System Conditions", the step to ensure no compensatory actions are in effect which degrade the Control Room Pressure Boundary during performance of Enclosures 13.9 through 13.17 is being replaced with the following steps:

   a) Ensure at least one train of the Control Room Chilled Water/ Ventilation System is OPERABLE.
   b) Ensure that neither train of the Control Room Ventilation System is open for maintenance or modification activities (i.e. Control Room Area Pressurizing Filter Train, Control Room Air Handling Unit, or associated ductwork).
   c) Ensure that no maintenance or modifications are in effect which degrade the control room pressure boundary (i.e. doors, firestops, cable penetrations, etc.).

5) In the section about "Test Method", add the test acceptance criteria

drawing, "CNTC-1564-VE.SO04-01", to Steps 9.10,911, 9.14,9.15, and 9.17. In Section 11, "Acceptance Criteria", add "CNTC-1564-VE.SO04-01" to Step 11.5 for the alignments associated with the above steps. Add Note 1 under Step 11.5 to identify the remaining enclosures as troubleshooting tests.

6) In Enclosures 13.1, 13.5, 13.11, and 13.15, delete the step and note associated with the implementation of the VCNA compensatory action for control room ventilation with the Auxiliary Building Ventilation System in abnormal alignment and control room pressurization test within 30 minutes of placing the Auxiliary Building Ventilation System in its LOCA alignment.

7) In Enclosures 13.9 through 13.16, delete the step requiring annulus drawdown and Airflow tests to be performed before the Annulus Ventilation System train can be declared operable.

8. In Enclosures 13.11, 13.15, and 13.17, change the step associated with not meeting Acceptance Criteria 11.5 to notify the Unit/WCC SRO to log the "Reactor Building" instead of the subject Annulus Ventilation Train in TSAIL.

9) In Section 9, steps 9.9 and 9.13 change Enclosures 13.9 and 13.13 to troubleshooting tests. In Enclosures 13.9, 13.12, 13.13, and 13.16 steps associated with not meeting the acceptance criteria were changed to support these enclosures as only troubleshooting tests. The steps associated with not meeting the acceptance criteria were deleted.

10) In Section 9 steps 9.10, 9.11, 9.14, and 9.15, information regarding maintenance activities and declaring the Reactor Building inoperable was added.

11) In Enclosures 13.10 and 13.14, steps associated with not meeting the acceptance criteria were changed to support actions documented in CNC-1240.00-00-0009, Revision 2.

12) In Section 9, steps 9.9 through 9.17, Section 11, Acceptance Criteria 11.5, and the applicable steps in Enclosures 13.9 through 13.17 (including data sheets), change the annulus pressure data points from -3.45 and -0.505 to -3.5 and -0.50 inwg.

13) In Section 9 steps 9. 1 and 9.5, Section 11 Acceptance Criteria 11. 1, and Enclosures 13.1 and 13.5 step 1.6, change -1.25 to -1.20 inwg.

14) In Section 9 step 9.18, Section 11 Acceptance Criteria 11.6, Enclosures 13.1 and 13.5 step 1.14, and Enclosure 13.18, steps 1.7, 1.8, and 1.9, change the annulus pressure stabilization range from -1.25 and -1.75 to -1.20 and -1.65 inwg.

15) In Section 11, change Acceptance Criterion 11.5 from 84 to 87 seconds.

16) Minor editorial changes are being incorporated in Sections 1, 9, 11, and 12. In Section 1 step 1. 5, revise this step to provide additional information regarding the justification and purpose for additional single train Annulus Vacuum Decay Testing. In Section 1, step 1.2, Section 9, steps 9.2 and 9.6, and Section 11, step 11.2, change the word "charcoal" to "carbon". In Section 11, step 11.5, added a note to identify each troubleshooting enclosure. In Section 12, the Notes were revised to support changes below for Enclosures 13.9 and 13.13 as only troubleshooting tests.

Evaluation: Reactor Building Operability is verified after refueling outages prior to Mode 4, "Unit Startup". Operability of the Annulus Ventilation System will be restored in the unlikely event that a safety injection occurs during the vacuum decay tests with the Unit in Modes 1 through 4. The ability of the Annulus Ventilation System to perform its design basis functions during a design basis accident will not be degraded. Offsite and Operator doses will remain within 10CFR100 and GDC 19 limits. Therefore, the margin of safety as defined in the basis for any Technical Specification will not be reduced.

The major procedure changes described herein will not adversely affect the technical aspects of the annulus vacuum decay tests. Items 1 through 3 support testing without any compensatory actions. Items 4 and 6 support removal of compensatory actions from the procedure due to NRC compensatory action concerns and being unnecessary. Item 7 removed unnecessary testing formerly required after some vacuum decay tests. Error uncertainties were removed from these procedures because they weree determined to be unnecessary. Changes associated with CNC-1240.00-000009, Revision 2 were incorporated into these procedures. Other minor changes were incorporated. The test methods and intent of the procedures did not change.

There are no unreviewed safety questions associated with this procedure revision. No Technical Specification changes are required. No UFSAR changes are required..

189 **Type:** Procedure **Unit:** 1

Title: Procedure PT/1/A/4700/012 "Standby Shutdown Facility Control Panel Functional Verification", Revision 3A

Description: This procedure restricted change allows procedure PT/1/A/4700/012 "Standby Shutdown Facility Control Panel Functional Verification" to proceed while valves 1SA-145, 1 CA-178, 1NV-876, and 1NV-877 are unavailable.

For valves 1SA-145 and 1CA-178, the transfer relays in electrical cabinet 1ELCC0034 will be verified to have energized when control is transferred to the Standby Shutdown Facility and verified to have de-energized when control is returned to the control room. Upon completion of the work on these valves, they will be stroked from the SSF and the method documented on a procedure discrepancy.

Valves 1NV-876 and 1NV-877 can only be controlled from the Standby Shutdown Facility. Upon completion of the work on these valves, they will be stroked from the Standby Shutdown Facility and the method documented on a procedure discrepancy.

Evaluation: This test can only be performed while Unit 1 is in Mode 5, 6, or No Mode. During these modes, the equipment affected by these valves (Turbine Driven Auxiliary Feedwater Pump for 1SA-145 and 1CA-178, and the Standby Makeup Pump for 1NV-876 and 1NV-877) is not required to be operable.

No unreviewed safety questions are created as a result of this procedure revision because no analyzed accident could be initiated by this test while Unit 1 is in Mode 5, Mode 6 or No Mode. No Technical Specification changes or FSAR changes are required for the procedure change,

224 **Type:** Procedure **Unit:** 1

Title: Procedure PT/1/A/4700/012, "Stanby Shutdown Facility Control Panel Functional Verification Test" Revision 3B

Description: Procedure PT/1/A/4700/012 Revision 3B deletes the pre-requisite that the valves listed in procedure step 8.3 must be able to be stroked. This function was previously verified during refueling outage 1EOC12 when performing Enclosures 13.2, 13.3, 13.4, 13.5, and 13.7. Enclosure 13.1 was not performed with the other enclosures because valve 1SA-145 was disassembled and valve 1CA-178 was a block tagout boundary. Thus, it is not necessary to stroke the valves in step 8.3 to perform this enclosure.

Evaluation: This procedure is only performed in Mode 5 or below, when neither the Auxiliary Feedwater Pump Turbine nor the Standby Shutdown Facility is required to be operable. This change does not affect the test method nor does it change the acceptance criteria. There are no unreviewed safety questions associated with this procedure revision. No Technical Specification changes are required. No UFSAR changes are required.

102 **Type:** Procedure **Unit:** 2

Title: Procedure PT/2/A/4200/01N, " Reactor Coolant System Pressure Boundary Leak Rate Test", Revision 39

Description: The purpose of this procedure is to verify that the leakage past any Reactor Coolant System Pressure Isolation Valve (PIV) does not exceed the value specified in Tech Spec 3.4.14 by satisfying the surveillance requirement of Technical Specification Surveillance Requirement 3.4.14.1. The valves that are tested by this procedure are important in preventing over pressurization and rupture of the ECCS low pressure piping which could result in a Loss of Coolant Accident (LOCA) that bypasses containment. The valves tested are the first and second stage check valves in the Safety Injection cold leg and hot leg injection lines, as well as the Residual Heat Removal suction isolation gate valves off the B and C hot legs.

The purpose of this change is to retype the procedure to incorporate procedure changes previously approved as Revision 38A and 38B, and the additional changes as described below.

1. A "Prerequisite System Condition" step has been added to ensure all lights on the Safety Injection Test Panel are working properly. This step provides additional assurance that the test panel will provide valve position indication to support the test, and prevents potential test delays.

2. On Enclosure 13.31 and 13.32, the PIV test pressure has been reduced. During the last revision, the test pressure was restricted to not less than 100 psig, due to the test being performed in Mode 4 (with Reactor Coolant System temperature less than 260 degrees F. T avg). At the maximum system temperature, flashing can occur across the Residual Heat Removal isolation valves or in the test rig if the test pressure drops below 21 psig (P sat). After accounting for elevation differences, this revision will restrict test pressure to not less than 40 psig at the test rig and not less than 70 psig at the Residual Heat Removal pump suction. This test pressure provides adequate margin to prevent flashing while achieving a maximum pressure drop across the Residual Heat Removal isolation valve, which provides additional assurance that the PIV leakage test will produce accurate leakage data.

3.On Enclosure 13.31 and 13.32, a "Prerequisite System Condition" note was revised to allow performance of step 8.1 through 8.5 in any order at the discretion of the test coordinator. By including step 8.5, the test instruments may be recorded prior to pre-test notifications and approvals.

4.On Enclosure 13.31 and 13.32, step 12.4 was revised and new step 12.30 has been added. These steps verify a stable or slightly increasing Residual Heat Removal pump suction pressure. Any significant change in Residual Heat Removal suction pressure would indicate a change in system condition that could affect test results.

5.On Enclosure 13.31 and 13.32, a note has been added after step 12.4 for removal of the Residual Heat Removal pump suction venting rig. Once a stable Residual Heat Removal suction pressure has been achieved, the note allows performing step 12.35 at any time at

the discretion of the test coordinator.

6.On Enclosure 13.31 and 13.32, step 12.25 has been revised to maintain approximately equal pressures at the "Test Gauge" and downstream of the second loop isolation valve. The procedure originally required maintaining pressure within 1 psig, which was difficult to achieve and unnecessarily restrictive. By maintaining an approximately equal pressure, leakage past the second loop isolation valve will be minimal and insignificant as compared to leakage past the first loop isolation valve (which will have a differential pressure of approximately 250 psi).

7.On Enclosure 13.31 and 13.32, new step (in section 12.41) adds instructions to return breakers for the Reactor Coolant System loop isolation valves to the "ON" position. Adding this step improves procedure progression.

8. On Enclosure 13.31 and 13.32, step 12.41.4 has been revised to "bullets" format. By using "bullets", these steps will be performed in any order as determined by the control room operator (OATC). Since this step has confirmed that a subcooled condition will be maintained, specifying a sequence for reopening these valves is not required.

9. On Enclosure 13.33, the location information about valve 2NI474 has been corrected.

10. On Enclosure 13.38, the test sequence has been revised to complete Enclosure 13.27 prior to other PIV test. The new sequence will test the first check valves off the Reactor Coolant System hot leg prior to testing the second check valves. This sequence will result in a higher pressure between the two check valves at the completion of the PIV test, which improves valves performance during the non outage periods.

11. Various document format changes have been incorporated into this procedure change. These do not affect the procedure's technical content or test method, and therefore are not individually identified.

Evaluation: UFSAR documents do not include any details regarding the method for performing the Reactor Coolant System PIV test. The PIV test method is not changing as result of this procedure change. A significant portion of these changes are administrative in nature or are being added for clarification. These do not affect test method or results. The change in test sequence on Enclosures 13.38 also does not impact the test method, but it does provide an improvement in post-test valve performance, which is considered an enhancement.

This procedure change will improve the PIV test by adding verification that the Residual Heat Removal Pump suction pressure is stable or increasing slightly. A significant change in Residual Heat Removal Pump suction pressure could indicate a change in system condition that could affect test results (i.e., leakage into the suction header, leakage from vent and drain valves, or a change in system alignment). Also for testing Residual Heat Removal suction isolation valves, the revised procedure requires an approximately equal pressure at the inlet and outlet of the second valve off the Reactor Coolant System loop. The procedure originally required maintaining pressure within one psig, which was difficult to achieve and unnecessarily restrictive. Prior to these steps, the PIV test has verified that the first loop isolation valve is leaking greater than the second loop isolation. Therefore, by maintaining a stable or slightly increasing pump suction pressure and

approximately equal pressures across the valve, leakage past the second loop isolation valve is considered minimal and insignificant as compared to leakage past the first loop isolation valve (which will have a differential pressure of approximately 250 psi). Together, these changes provide an enhancement to the PIV leakage test.

This procedure change will lower the required PIV test pressure for Residual Heat Removal Pump suction isolation valves. In the last procedure revision, the test pressure was restricted to not less than 100 psig, due to the test being preformed in Mode 4 (with the Reactor Coolant System temperature less than 260 degrees F. T avg). At the maximum system temperature, flashing can occur across the Residual Heat Removal isolation valves or in the test rig if the test pressure drops below 21 psig (Psat). After accounting for elevation differences, this revision will restrict test pressure to not less than 40 psig at the test rig and not less than 70 psig at the Residual Heat Removal Pump suction. This test pressure provides adequate margin to prevent flashing while achieving a maximum pressure drop across the Residual Heat Removal isolation valve, which providing additional assurance that the PIV leakage test will produce accurate leakage data.

UFSAR Section 6.3.4.2 "Reliability Tests and Inspections" describes the purpose of the Reactor Coolant System PIV (i.e., PBV -Pressure Boundary Valve) leak test and describes the Safety Injection test header. Neither the method for performing the test or specific components used in the test are discussed.

SER 3.9.6, Inservice Testing of Pumps and Valves, describes the requirements for establishing the Reactor Coolant System PIV leakage limits and test. The SER describes that a valve leaking greater than one gpm may be an indication of degradation; thus the administrative limit for any single valve shall be one gpm. The maximum allowable leakage for each valve is identified by Technical Specification 3.4.14.

These changes to the test procedure do not affect the reliability of the isolation function, violate any licensing or design basis, or increase the probability or consequences of any design basis event.

The changes presented during this re-type of the procedure do not contradict any information presented in the UFSAR.

The Chapter 15 Accident Analyses were reviewed and are not affected since this test is performed within the parameters of the system's design function. The method of testing these valves has not changed, with the overall effect being enhancement to the test process. The test acceptance criteria will remain the same. Since this test will be performed using the slightly enhanced process and bounded by the same acceptance criteria, the probability of occurrence of an accident previously evaluated in the SAR will not increase.

The Safety Injection and Residual Heat Removal Systems are aligned per valve lineup and as specified in this procedure. The valve lineup along with valve manipulation in each enclosure ensures that (a) a valid test flow path exists, (b) each valve under test is pressurized, and (c) no direct flow path from the Reactor Coolant System to the auxiliary building exists. All valves will be aligned and operated per administrative guidelines (procedure); therefore, the probability of occurrence of a malfunction of equipment

important to safety previously evaluated in the UFSAR is not increased.

During this test, the alignment is such that a direct flow path will not be created from the Reactor Coolant System to the auxiliary building (inter-system LOCA). All valves are assured of being tested and meeting their required leakage. Therefore, the consequences of an accident previously evaluated in the UFSAR are not increased.

All valves are aligned and operated per administrative guidelines (procedures); therefore, the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR are not increased.

The test does not involve any hardware modifications or operation outside normal limits. No system will be put in a jeopardized configuration outside the allowable configuration of Technical Specifications. The test flow path is via two containment isolation valves (2NI-95A and 2NI-96B). In the event power is lost or in the event of a LOCA, the flow path will be isolated by closing one of the isolation valves. Therefore, the possibility for an accident of a different type than any previously evaluated in the UFSAR is not created. This test does not involve any hardware modifications or operation outside normal limits. Therefore this test does not adversely affect any equipment important to safety. There are no new malfunctions of equipment important to safety different than any already evaluated previously in the UFSAR.

An administrative limit of "less than one gallon per minute leakage" is applied to all enclosures. This meets and for most PIV's exceeds , the maximum allowable leakage allowed by the Technical Specifications. Therefore the margin of safety as defined by Technical Specifications is not being affected by the performance of this procedure with these changes incorporated.

There are no unreviewed safety questions associated with this procedure change. No Technical Specification changes are required. No UFSAR changes are required.

1 **Type:** Procedure **Unit:** 2

Title: Procedure PT/2/A/4200/081 Revision 0, Valve 2NV-849 Travel Stop Verification

Description: Procedure PT/2/A/4200/018 (2NV-849 Travel Stop Verification) was developed to ensure that the travel stops to limit the maximum allowable flow are set properly following maintenance on valve 2NV-849 (Normal Letdown Variable Orifice). This procedure will be run when the Reactor Coolant System is at normal operating temperature and pressure. Valve 2NV-849 will then be placed in service by itself. Normal letdown flow will be increased until valve 2NV-849 is either wide open or a flow rate of 110 gpm is reached, whichever comes first. Maintenance will then adjust the travel stops on the valve such that 110 gpm is the maximum allowable flow. Normal letdown flow is then reduced as desired. The procedure will then allow Operations to restore the desired alignment of the letdown orifices.

Catawba Nuclear Station discontinued using the Chemical and Volume Control System variable letdown orifice valves NV-849 upon discovering that the valve and associated piping were experiencing high vibration.

Valve 2NV-849 had been qualified following the identification of the vibration problem for short term use to warm the letdown line in the case of initiating letdown or the re-initiation of letdown following a loss of letdown. The valve was to be used for this purpose only when Reactor Coolant System pressure is greater than 350 psig. When the Reactor Coolant System was less than 350 psig, valve 2NV-849 could be used at all times.

Subsequently the piping was reconfigured using butt welds instead of the originally installed socket welds, which significantly reduced the vibration in the piping. Valve 2NV-849 trim cage was replaced with a newer version less susceptible to producing vibration. The letdown line was retested and the vibration levels measured. The Unit 2 data gathered supports unconditional use of valve 2NV-849 for any allowable flow conditions using any combination of different orifices. Unit 2 was returned to a condition within the original design of the letdown line. Valve 2NV-849 can be placed in service for as long as desired. The travel stops for valve 2NV-849 are to be set up for a maximum allowable flow of 110 gpm at normal Reactor Coolant System operating temperature and pressure. Valve 2NV-849 is now used as originally designed and described in the UFSAR.

Evaluation: Valve 2NV849 is not safety-related nor is it used to mitigate any of the accident scenarios described in the UFSAR. The flow path through valve 2NV-849 is isolated by safety-related equipment that operates on safety-related signals.

Valve 2NV-849 will only be operated within the confines of the already approved operating procedure, OP/2/A/6200/001 (Chemical and Volume Control System). The operating procedure operates the valve as originally designed. There will be no operations with 2NV-849 outside of the bounds addressed in the UFSAR.

There are no unreviewed safety questions associated with this procedure. No Technical Specification changes are required. No UFSAR changes are required.

200 **Type:** Procedure **Unit:** 2

Title: Procedure PT/2/A/4200/13H, NI and NV Check Valve Test Revision 15

Description: This revision to the NI/NV (Safety Injection System/Chemical and Volume Control System) check valve test is being made to add steps that verify full stroke of the two inch Safety Injection System discharge pressure boundary valves (NI-165, NI-167, NI-169, and NI-171) and the common Safety Injection System pump suction check valve from the Refueling Water Storage Tank (NI-101). The IST full stroke requirement for these valves was met in the past by performing PT/2/A/4400/01, "ECCS Flow Balance". Catawba will no longer perform a full flow balance verification during outages if no work was done that would require this test. The procedure has guidance in it to ensure that the Operations locked valve verification procedure has been performed prior to this test. This prerequisite will assure the ECCS throttle valve positions have not been adjusted from the previous set position. Technical Specification Surveillance Requirement (SR) 3.5.2.7 requires that the position stops for the ECCS throttle valves are in their correct position on an 18 month frequency. This surveillance is met by the verification of mechanical locks and tamper seals for each ECCS throttle valve. A review of the Technical Specification Bases for this Surveillance Requirement indicates the need for mechanical locks, but does not require an actual flow balance to prove correct throttle valve position.

If a full flow balance is not performed each outage, there is a need to move the IST stroke requirement to the NI/NV Check valve test. Also, a less stringent requirement on cavity water level has been written into the Limits and Precautions section. Previously, the requirement on maximum cavity level was 87%. The new requirement is 94% Reactor Coolant System wide range level. This new maximum is still well below the point where water would spill over the reactor cavity windows and is consistent with the Technical Specification required level for refueling operation.

Because flow balance will not be verified, steps have been added to record pump head data at the balance flow point for the Safety Injection and Centrifugal Charging Pumps. The collection of head data requires more test instrumentation than what was previously included in the NI/NV check valve test procedure. Steps have been added in this retype to ensure that the proper instruments are installed and removed. The instruments are consistent with those required by procedure PT/2/A/4400/01. Certain "Caution" statements regarding valve position have been removed from Section 8.0 because the procedure and valve lineup adequately control configuration. In addition, several "Caution" statements were removed from Section 12.

Certain motor-operated valves are periodically required to have static and DP testing. Steps were added to perform this for valves 2NI-333B, 2NI-332A, 2NI-334B, and 2NV-312A. Although none of these valves will be tested during the 2EOC1O refueling outage, testing steps were added for these valves to make the Unit 2 procedure more consistent with the Unit 1 procedure. The steps were made conditional so that they could be marked as "Not Applicable" if these valves are not on the test list for future outages.

The purpose of PT/2/A/4200/13H, "NI/NV Check Valve Test", is to comply with Catawba IST program requirements for operability (full and partial stroke exercise) for those valves listed in the procedure. Including additional valves in the test list adds a new

flow path (Safety Injection System to Reactor Coolant System Cold Legs) to the procedure, but the justification for adding this flow path is no different from the Justification for allowing the other parts of the test. The reactor vessel is open with no fuel in the core during performance of this test. Safety Injection and Centrifugal Charging pumps discharge into the reactor vessel and water is allowed to overflow into the reactor vessel cavity. Since the testing is performed while the Unit is in "No Mode", none of the ECCS systems are required to be operable during this time. Limits and precautions are in place to ensure that the pumps are protected from runout. Flow below the runout limit is assured by throttle valve position that will not have changed since the last full ECCS flow balance.

The new steps for process instrumentation installation and restoration serve to ensure that those instruments important to plant operation are returned to a functional state after removal of test instrumentation. These steps involve no physical changes to the plant beyond that already required in the procedure, and act as a second check that the process instrumentation has been returned to service. The ECCS systems are not required to be operable in "No Mode" so the act of installing test instrumentation will not affect the operability of any ECCS component. Removal of "Caution" statements in the Prerequisite section and the procedure body result in editorial changes to the procedure. In each case, configuration was adequately controlled by valve lineup. Also, operator training emphasizes that pumps must have both a suction source and a discharge pathway prior to being started.

Evaluation: There are no unreviewed safety questions associated with this procedure revision. The testing is performed in "No Mode" with no fuel in the reactor vessel. Therefore there is no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required

13 Type: Procedure **Unit:** 2

Title: Procedure PT/2/A/4400/01 Revision 29, "ECCS Flow Balance"

Description: Revision 29 to Procedure PT/2/A/4400/01 makes changes to the existing acceptance criteria for each section. These changes are required due to the recent completion of an error analysis calculation for flow balance testing and pump head curve verification. Several other minor editorial changes were made. The editorial changes do not significantly affect the test procedure or the data/results obtained from the test.

Evaluation: The procedure changes made to the acceptance criteria for procedure PT/2/A/4400/01 have no adverse impact on plant safety. This test is performed with no fuel in the core. The purpose of the test as described below is still applicable and the intent of the test is not compromised as a result of these procedure changes. The ECCS equipment will be operated within the specified design limits and criteria.

The Emergency Core Cooling System is a functional grouping of the following systems or portions thereof: Refueling Water, Residual Heat Removal, Chemical and Volume Control, and Safety Injection. The purpose of the ECCS is to provide emergency cooling to the reactor core in the event of a Loss of Coolant Accident (LOCA) or steam line break.

Test Acceptance Criteria (TAC) sheet requirements for flow balance testing and pump head curve verification provide assurance that proper ECCS flow will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to:

1) Prevent pump flow from exceeding runout limits;
2) Provide the proper flow split between injection points; and
3) Provide an acceptable level of total ECCS flow to all injection points equal to or above ECCS LOCA analysis assumptions.

Performance of PT/2/A/4400/01, ECCS Flow Balance, ensures that Safety Analysis assumptions regarding pump runout, total flow, and flow split requirements are satisfied and that the ECCS flow resistance and pressure drop characteristics are adequate for the system to perform its intended safety function. Accident analysis assumptions (in the form of TAC sheet requirements) must be adjusted for instrument error. A formal calculation documenting the error associated with each tested parameter has been recently completed. The new calculation requires that adjustments be made to the acceptance criteria found in PT/2/A/4400/01. Adjusting the TAC sheet requirements in this way assures that the accident analysis assumptions remain bounding.

There are no unreviewed safety questions associated with this procedure change. The ECCS flow balance is performed in "No Mode" with no fuel in the reactor core. No ECCS component is an accident initiator. No Technical Specification changes are required. No UFSAR changes are required.

247 **Type:** Procedure **Unit:** 2

Title: Procedure PT/2/A/4400/013 "Component Cooling System Pump Interaction Test"

Description: Procedure PT/2/A/4400/013 "Component Cooling System Pump Interaction Test" is being deleted. The procedure was used in testing the miniflow characteristics of the Component Cooling Pumps.

The Catawba Component Cooling System acts as a safety related closed loop treated water system to dissipate waste heat from motor coolers and intersystem heat exchangers serving various systems. In addition, the Component Cooling System provides cooling to engineered safeguards loads after a Design Basis Event. This system serves as a boundary between the Reactor Coolant System and the Nuclear Service Water System; reducing the probability of radioactivity leakage into the environment.

NRC IE Bulletin 88-04 requested that all licensees investigate and correct two miniflow design concerns. The first involves the potential for dead-heading one or more pumps in safety-related systems that have a miniflow line common to two or more pumps of other configurations that do not preclude pump-to-pump interaction during miniflow operation. The second issue is whether or not the installed miniflow capacity is adequate for even a single pump in operation.

Included in the Catawba response to Bulletin 88-04 was a commitment to perform detailed calculations to identify pump flows resulting from appropriate valve alignments while imposing IWP acceptance criteria, to investigate short term minimum flow requirements from the manufacturer, and to evaluate possible setpoint changes to preclude unacceptable pump interaction.

As a result of these commitments, the minimum flow setpoints were changed and the procedure PT/2/A/4400/013 "Component Cooling System Pump Interaction Test" was developed to verify the new setpoints. This procedure failed to produce adequate results and was not able to verify that no interaction existed. Instead of performing this test, analyses have been performed in association with test procedure PT/2/A/4400/03F, "Head Curve Test for Component Cooling System Pumps 2A1, 2A2, 2B1, and 2B2" to ensure a strong pump/weak pump interaction does not exist. The pump head curve tests and Interaction Analysis are adequate to meet the Catawba IE Bulletin 88-04 commitment.

Evaluation: There are no unreviewed safety questions associated with deletion of this procedure. The probability of an accident previously evaluated in the UFSAR will not be increased. The Component Cooling System is a safety related system used to mitigate design basis events. It is not an accident initiator and therefore, deleting the procedure will not increase the probability of an accident previously evaluated in the UFSAR. Deleting the test procedure is an administrative change that does not affect the operation of the system.

The probability of a malfunction of equipment important to safety will not be increased. The procedure being deleted was ineffective in determining whether a strong pump/weak pump interaction exists. Deleting the procedure will therefore not increase the probability of a malfunction. Strong pump/weak pump interaction has been analytically evaluated in several engineering calculations. Where it was found that interaction potential exists, administrative controls have been put in place to prevent the alignments from occurring.

In general, the safety related Component Cooling Pump minimum flow line and control valve provides adequate protection for all credible Component Cooling System alignments during normal operation and design basis response to accidents. Administrative Controls are in place to avoid system alignments where a strong pump/weak pump interaction may occur in excess of that protection provided by the minimum flow path. No Technical Specification changes are required. No UFSAR changes are required.

183 **Type:** Procedure **Unit:** 2

Title: Procedure PT/2/A/4400/014, "Nuclear Service Water System to Auxiliary Feedwater System Suction Piping Flow Measurement", Revision 4A

Description: A restricted change (Revision 4A) was made to procedure PT/2/A/4400/014, "Nuclear Service Water System to Auxiliary Feedwater System Suction Piping Flow Measurement". These restricted changes only apply when:
Unit 1 is shut down (Mode 5, 6 or No-Mode)
Unit 2 is at power
Nuclear Service Water System to Auxiliary Feedwater System flow testing is only performed on the inoperable Nuclear Service Water System train
Nuclear Service Water System trains are separated

Evaluation: The Nuclear Service Water System including Lake Wylie and the Standby Nuclear Service Water Pond (SNSWP), is the ultimate heat sink for various nuclear safety related heat loads during normal operation and design basis events. The Nuclear Service Water System supports Emergency Core Heat Removal (ECCS) operation by providing cooling to the Component Cooling System via the Component Cooling System heat exchangers and also to the Diesel Generators via the Diesel Generator Engine Jacket Water Cooler system heat exchangers. Other nuclear safety related loads include the Containment Spray heat exchanger and Control Room Chilled Water System Chiller Condenser. The Nuclear Service Water System also provides assured makeup to the Component Cooling System, Spent Fuel Pool, Auxiliary Feedwater supply and the Containment Seal Water Injection System.

The Nuclear Service Water System shall be capable of mitigating the consequences of a design basis event on one unit concurrent with a Loss of Non-Emergency A/C Power (loss of offsite power) affecting both units, assuming a single active failure. The most stringent design basis event for the Nuclear Service Water System is a Loss of Coolant Accident (LOCA) on one unit. The following assumptions are further postulated for the Nuclear Service Water System design basis LOCA:

Normal Shutdown of remaining unit from normal operation
Prolonged Drought in hot weather (maximum supply temperature/minimum supply volume)
Loss of Lake Wylie

Restricted Change (Revision 4A) to PT/2/A14400/014, Nuclear Service Water System to Auxiliary Feedwater System Suction Piping Flow Measurement deletes the prerequisite conditions to require four operable Nuclear Service Water System pumps and four Operable D/Gs, and adds steps to ensure that Unit 1 is shut down (Mode 5, 6 or No-Mode), Unit 2 is at power, Nuclear Service Water System to Auxiliary Feedwater System flow testing is only performed on the inoperable Nuclear Service Water System train, and the Nuclear Service Water System trains are separated. The entire train of Nuclear Service Water associated with the train being tested is considered inoperable. The Operable train is unaffected by the Nuclear Service Water System to Auxiliary Feedwater System flow test on the inoperable train, since the trains will be separated. The Operable Nuclear Service Water System train is bounded by the Nuclear Service Water System one pump flow balance. The requirement for four operable Nuclear Service Water System

pumps and D/Gs is not applicable to the inoperable train, therefore these requirements can be deleted when:

Unit 1 is shut down (mode 5, 6 or no-mode), Unit 2 is at power, Nuclear Service Water System to Auxiliary Feedwater System flow testing is only performed on the inoperable Nuclear Service Water System train, and Nuclear Service Water System trains are separated.

There is no unreviewed safety question associated with this procedure revision. No Technical Specification changes are required. No UFSAR changes are required.

142 **Type:** Procedure **Unit:** 0

Title: Procedure TN/1/A/1554/MI/002 "1RN-003A Partial Stroke Troubleshooting Plan"

Description: The A Train Nuclear Service Water System Pumps are currently aligned to the Standby Nuclear Service Water Pond per Removal and Restoration Sheet 00-1793, "Nuclear Service Water System Alignment to Maintain Train A Operability". Per procedure TN/1/A/1554/MI/002 "1RN-003A Partial Stroke Troubleshooting Plan", the white tag for valve 1RN-003A will be removed and it will be partially stroked closed. Per this procedure, the manual valve strokes will maintain the valve greater than or equal to 50 degrees open at all times, so the valve will pass more flow than the 40 degrees open that is described in this evaluation. The Train A Nuclear Service Water Pumps will not be running during this test, but they will be capable of starting. Upon completion of the procedure, the valve will be returned to the full open position and white tagged. For 1RN-003A valve position greater than 40 degrees open, flow rates through the valve are sufficient to maintain Nuclear Service Water System operability.

The Nuclear Service Water System, including Lake Wylie and the Standby Nuclear Service Water Pond, is the ultimate heat sink for various nuclear safety related heat loads during normal operation and design basis events. The Nuclear Service Water System supports Emergency Core Heat Removal operation by providing cooling to the Component Cooling System via the Component Cooling System heat exchangers and also to the Diesel Generators via the Diesel Generator Engine Jacket Water Cooler System Heat Exchangers. Other nuclear safety related loads include the Containment Spray Heat Exchanger and Control Room Chiller Condenser. The Nuclear Service Water System also provides assured makeup to Component Cooling, Spent Fuel Pool, Auxiliary Feedwater supply and the Containment Seal Water Injection System.

The Nuclear Service Water System shall be capable of mitigating the consequences of a design basis event on one unit concurrent with a loss of offsite power on both units, assuming a single failure. The most stringent design basis event for the Nuclear Service Water System is a Loss of Coolant Accident on one unit. The following assumptions are further postulated for the Nuclear Service Water System design basis LOCA:

1. Normal Shutdown of remaining unit from normal operation
2. Prolonged Drought in hot weather (maximum supply temperature/ minimum supply volume)
3. Loss of Lake Wylie

The Nuclear Service Water System Design Basis Document states that:
Valve 1RN-003A is normally closed to isolate the Nuclear Service Water System from the Standby Nuclear Service Water Pond. The valve opens on a 2/3 emergency low level in either Nuclear Service Water System Pump House Pit to align the Standby Nuclear Service Water Pond to the Nuclear Service Water System Pump Pit A. Nuclear Service Water Pumps 1A and 2A draw suction from intake pit A. Valve 1RN-003A is controlled from pushbuttons in the Control Room on main control boards 1MC11 and 2MC11. Valve 1RN-003A will open automatically when the transfer switch on the Unit 1 A Auxiliary Shutdown Panel is closed and at least one Nuclear Service Water Train A pump is running.

The design basis document recommends the following action statement:

With valve 1R-00N3A closed and incapable of automatically opening, the A loop of the Nuclear Service Water System is inoperable. There is no recovery action to maintain operability in this scenario. With valve 1RN-003A open and incapable of closing, the operability of the A loop is maintained, but normal operating procedures would be affected.

A referenced note states: With valve 1RN-003A open and incapable of closing, the A loop of the Nuclear Service Water System (suction and discharge) should be aligned to the the Standby Nuclear Service Water Pond to prevent loss of the Standby Nuclear Service Water Pond inventory to the lake. Also, operability of the the Standby Nuclear Service Water Pond may be affected under Technical Specification 3.7.9.

Evaluation: This temporary procedure alters the Nuclear Service Water System by allowing partial closing of valve 1RN-003A, the A Train Nuclear Service Water System pumps inlet isolation valve from the the Standby Nuclear Service Water Pond. The Operability of Nuclear Service Water System Train A is maintained as long as the valve is maintained greater than 40 degrees open. At 40 degrees open with an assumed Nuclear Service Water System flow from the A Pit of 25,000 gpm, the piping resistance changes will cause the Nuclear Service Water System A Pit level to decrease by approximately 0.6 feet, in addition to the 1.5 feet described in a previous 10CFR50.59 evaluation about valve 1RN-003A. This additional 0.6 foot change in pit level is well within all valve and pit level design parameters.

Therefore, the change in piping resistance caused by closing the valve 50 degrees from the full open position would cause level in the Nuclear Service Water System Pit A to decrease by an additional 0.6 feet with a assumed flow rate of 25,000 gpm. Lower flow rates would cause a smaller change in pit level.

There is no Unreviewed Safety Question associated with procedure TN/1/A/1554/MI/002. No Technical Specification changes are required. No UFSAR changes are required.

8 **Type:** Procedure **Unit:** 2

Title: Procedure TO/2/A/6200/001 Revision 0 and OP/2/A/6150/006 Revision 55D

Description: During the 2EOC10 Refueling Outage a problem developed with obtaining the required amount of seating surface for valve 2NI-71. Another work window was required to complete the work on the valve. The work window was after fuel reload when the Reactor Coolant System was drained to below the flange level. Since valve 2NI-71 is the primary check valve off the Reactor Coolant System for cold leg injection into the "B" cold leg. A large vent path was established to lessen the change of core uncovery due to pressurization inside the vessel. Reactor Coolant System level was maintained below the level of the 2NC-71 check valve.

Evaluation: These procedure changes do not involve an unreviewed safety question. No Technical Specification changes are required. No UFSAR changes are required. Both trains of Residual heat removal being tested are capable of maintaining their residual heat removal function. Valve 2NI-71 can be repaired in Mode 5 or 6. There are no changes to any design limit or setpoint. No control, instrument function, or performance of any structure , system, or component is degraded.

167 **Type:** Procedure **Unit:** 1

Title: Procedure TT/1/A/9100/073, Safety Injection System Check Valve Acoustic Flow Test

Description: Temporary test procedure TT/l/A/9100/073, Safety Injection System Check Valve Acoustic Flow Test, configures the Residual Heat Removal System such that one pump is maintained in operation to cool the core, provide Residual Heat Removal System letdown and supply Pressurizer Auxiliary Spray if required. The alternate Residual Heat Removal System train is operated with the Heat Exchanger bypassed at various flow rates to allow non intrusive diagnostic testing of the six inch and ten inch Safety Injection System check valves. The test is scheduled to be performed in mode 5 and mode 6 while the Reactor Coolant System is not filled in mode 5 and less than 23 feet in mode 6. The test procedure configures the Residual Heat Removal System to vary flow through the tested train. Both Residual Heat Removal System trains are maintained operable during the performance of this test.

Evaluation: There are no unreviewed safety questions involved with this test procedure. No Technical Specification changes are required. No UFSAR changes are required.

30 **Type:** Procedure **Unit:** 0

Title: Procedures OP/1/A/6150/006 Revision 57B and OP/2/A/6150/006 Revision 55B "Reactor Coolant System Draining"

Description: Procedures OP/1/A/6150/006 (Reactor Coolant System Draining), Revision 57B (Unit 1) and OP/2/A/6150/006 (Reactor Coolant System Draining) Revision 55B (Unit 2) are being revised. These procedure revisions change the method of injecting nitrogen into the Steam Generators for draining purposes. Originally this procedure directed the operators to inject nitrogen to all four Steam Generators simultaneously. However, experience has shown that the operators are not able to maintain the reactor coolant system level within the procedure range while injecting nitrogen. Based on experience gained in the last outage, this procedure has been changed to inject nitrogen into one Steam Generator at a time. Once the initial bulk of the water has been drained from the Steam Generator, the procedure will allow the operator to begin the nitrogen injection on the next Steam Generator. Past experience with this technique from other Steam Generator nitrogen injections has shown that the operator can maintain Reactor Coolant System level within the band given in the procedure.

As the level in the Reactor Coolant System is lowered to below the reactor vessel flange level, water in the Steam Generator tubes tends to remain in place because the tubes are not vented to atmosphere. In order to displace the water in the tubes, Catawba has developed a technique to inject nitrogen into the tubes thereby displacing the water. Once the Reactor Coolant System water level has been lowered to 18 - 24 % or 18 - 22% (for incore thimble work in conjunction with reactor vessel head removal), the Reactor Coolant System Nitrogen Rig is hooked up to the Reactor Coolant System Flow Transmitter for each loop. The Reactor Coolant System Nitrogen Rig uses nitrogen from either the Bulk Nitrogen System or the Cold Leg Accumulators. The rig is capable of transmitting a pre-selected flow at a predetermined pressure to each of the Reactor Coolant System Flow Transmitters. The injection flow path to each Steam Generator is independent of the other Steam Generators. As the nitrogen enters the Reactor Coolant System Flow Transmitter, it enters the elbow taps in the Reactor Coolant System. The nitrogen will then flow upwards into the Steam Generator and into the tubes. Calculation CNC-1223.03-00-0023 (Steam Generator Primary Side Volumes and Nitrogen Backfill Evaluation and Manway Closure Evaluation) determined the flow rate for the nitrogen and how long the operators are to maintain that flow rate. Experience has shown that there is a great deal of water drained during the initial part of nitrogen injection. Based on that fact, once the drain rate of the Steam Generator falls off, the operators are allowed to begin nitrogen injection of the next Steam Generator while completing the injection of the required nitrogen for the previous Steam Generator.

Other procedure changes are to add steps to document the beginning and ending of the nitrogen injection for each Steam Generator. Another minor change was made to have the operator ensure the supply and Steam Generator hoses are hooked up to the Reactor Coolant System Nitrogen Rig. The original wording directed the operator to hook up the hoses when in fact, Maintenance should have already hooked them up. Another minor change was to ensure that all reference to the rig for injecting the nitrogen is referred to as the "Reactor Coolant System Nitrogen Rig".

Evaluation: There is no unreviewed safety question associated with these procedure revisions. The

activities described in the procedures take place after the "high energy" potential of the Reactor Coolant System has been dissipated. Prior to draining the steam generators using nitrogen injection, operations procedures have directed precautions for entering the "loops not filled" condition. Since the Reactor Coolant System's heat removal method is already considered to be impaired, other means of heat removal have been ensured to be available. The types of accidents discussed in the UFSAR are a loss of the operating train of Residual Heat Removal, loss of power to a Residual Heat Removal Pump, and lowering the Reactor Coolant System level down too far and air binding the pump. Loss of a Residual Heat Removal pump in a "Loops not Filled" condition is minimized since the other train of Residual Heat Removal is required operable by Technical Specifications. Backup power must be supplied to the Residual Heat Removal system via an emergency diesel generator. Strict procedural controls are in place to control the lowering of the Reactor Coolant System level. The UFSAR discusses that air binding a pump is not necessarily pump damaging, but usually requires that the pump casing be re-vented before the pump can be returned to service. In the mean time, borated sources of water are required to provide makeup to the Reactor Coolant System. These borated water supplies are considered always available since any pumps have emergency diesel generator power or the water can be gravity fed provided a large enough vent path has been established. The vent path and borated water sources are governed by Selected Licensee Commitments and a Site Directive which addresses Unit Shutdown Configuration Control. Therefore, injection of nitrogen into the Steam Generators is not considered an accident initiator. There is no increase in the probability of occurrence of an accident previously evaluated in the UFSAR. If too much nitrogen were to be injected, the nitrogen bubbles would run along the hot leg piping to the reactor vessel. Once in the reactor vessel, it would escape into the containment atmosphere since the reactor head vent is already open. There is no way for the nitrogen to build up in the hot leg in such a manner that it would interfere with the suction piping to the Residual Heat Removal Pump. Bubbles from injecting too much nitrogen would not impede any injection flow whether by pump or gravity feed. Therefore, injection of nitrogen into the Steam Generators is not likely to interfere with any equipment necessary for the safety of the core. The probability of a malfunction of safety related equipment is not increased. No Technical Specifications changes are required. No UFSAR changes are required.

29 **Type:** Procedure **Unit:** 2

Title: Procedures OP/2/A/6200/001M Revision 13, "Chemical and Volume Control System Fill and Vent" and OP/2/A/6200/006M Revision 14A "Safety Injection System Drain, Fill, and Vent"

Description: Procedure OP/2/A/6200/001M Revision 13 (Chemical and Volume Control System Fill and Vent) and procedure OP/2/A/6200/006M Revision 14A (Safety Injection System Drain, Fill, and Vent) were revised to optimize the fill and vent operations associated with returning the Safety Injection System, Chemical and Volume Control System and Residual Heat Removal System to service following maintenance. The procedure revisions affect procedure sections which are performed in "No Mode". When the plant is in "No Mode", all fuel is in the Spent Fuel Pool and the Spent Fuel Pool Cooling System, Component Cooling System, and Nuclear Service Water System are in service to provide the cooling to Spent Fuel Pool.

Evaluation: There is no unreviewed safety question associated with these procedure revisions. The activities described by the procedures take place when the reactor is defueled. With the reactor core in the Spent Fuel Pool, normal cooling is through the Spent Fuel Pool Cooling System to Component Cooling System to the Nuclear Service Water System. The only interaction with the Safety Injection System, Residual Heat Removal System, and Chemical and Volume Control System is that Component Cooling System and Nuclear Service Water Systems provide cooling to the Motor Coolers. The Spent Fuel Cooling System does not depend on the Component Cooling System for Spent Fuel Pool Cooling. If Component Cooling to the Spent Fuel System is lost, boiling occurs in the Spent Fuel Pool with assured makeup from the Nuclear Service Water System. With the reactor core in the spent Fuel Pool , fill and vent of the Safety Injection System, Residual Heat Removal System, and Chemical and Volume Control System will not be an accident initiator. The fill and vent activity will have no effect on the probability or consequences of accidents evaluated in the UFSAR. No Technical Specifications changes are required. No UFSAR changes are required

28 **Type:** Procedure **Unit:** 0

Title: Procedures PT/1/A/4350/12A Revision 0B, PT/1/A/4350/12B Revision 1E, PT/2/A/4350/12A Revision 0B, PT/2/A/4350/12B Revision 0C

Description: The acceptance criteria in procedures PT/1/A/4350/12A Revision 0B, PT/1/A/4350/12B Revision 1E, PT/2/A/4350/12A Revision 0B, PT/2/A/4350/12B Revision 0C is being changed to reflect criteria in the correct revision of Regulatory Guide 1.9. The procedure is not being changed, just the acceptance criteria. The revision number of Regulatory Guide 1.9 listed in the bases of Improved Technical Specification section 3.8.1.9 was changed from revision 0 to revision 2. It was determined that Catawba Nuclear Station is licensed to Regulatory Guide 1.9 revision 2, December 1979. Improved Technical Specification Surveillance Requirement 3.8.1.9 contains data from RG 1.9 revision 2 but the reference for the bases mistakenly listed revision 0. UFSAR Sections 8.3.1.2.4 and 8.3.1.1.3.11 also contains information from revision 0 of Regulatory Guide 1.9 so these sections will be revised as well. Per review of NUREG 0954 Safety Evaluation Report related to the Operation of Catawba Nuclear Station, Unit 1and 2, February 1983, Catawba Nuclear Station is licensed to Regulatory Guide 1.9 revision 2. Therefore, it is acceptable to use the information contained in this revision to satisfy respective diesel generator requirements.

Evaluation: There are no unreviewed safety questions associated with changing the acceptance criteria in the Diesel Generator Governor and Voltage Regulator test based on the correct revision of Regulatory Guide 1.9. No Technical Specification changes are required. UFSAR Sections 8.3.1.2.4 and 8.3.1.1.3.11 will be revised.

199 **Type:** Removed From Report **Unit:** NA

Title: NA

Description: This record was deleted from the report because the evaluation was never approved.

Evaluation: NA

214 **Type:** Removed From Report **Unit:** NA

Title: NA

Description: This record was deleted from the report because the evaluation was never approved.

Evaluation: NA

41 **Type:** UFSAR Change **Unit:** 0

Title: Change to UFSAR Chapter 16 Section 9

Description: Chapter 16 of the UFSAR (Selected Licensee Commitments) will be revised to add a commitment to allow the tracking of the availability of the Drinking Water System for the purpose of supplying emergency cooling water to the Centrifugal Charging Pumps during a loss of Component Cooling Water event.

Evaluation: There are no unreviewed safety questions associated with this UFSAR change. Addition of this commitment will have no effect on the probability or consequences of an accident. No Technical Specification changes are required. Chapter 16 of the UFSAR will be revised.

45 **Type:** UFSAR Change **Unit:** 0

Title: Change to UFSAR Section 7.6.15.1, Description of Diesel Generator Fuel Oil System Instrumentation and Controls

Description: UFSAR Section 7.6.15 addresses "Diesel Generator Fuel Oil System Instrumentation and Controls", Section 7.6.15.1 "Description" lists eleven system parameters that are alarmed. One of these is "10. Fuel recirculating filter high differential pressure". This parameter is being deleted and the list of parameters is being renumbered.

Evaluation: A review of this UFSAR section concluded that changing the UFSAR text to reflect that the important parameters of the Emergency Diesel Engine Fuel Oil System which are monitored and alarmed on the Diesel Control Panels do not include the "Fuel Oil Recirculating Filter High Differential Pressure." Removing this item and renumbering the list of parameters listed in the UFSAR does not create an Unreviewed Safety Question. This correction does not affect the licensing or design bases nor does it affect the operation or safety function of any system, structure, or component. No Technical Specification changes are required. UFSAR Section 7.6.15.1 will be revised.

46 **Type:** UFSAR Change **Unit:** 0

Title: Change to UFSAR Section 7.6.7.1, Description of Diesel Generator Room Sump Pump System

Description: UFSAR Section 7.6.7 addresses the "Diesel Generator Room Sump Pump System." Section 7.6.7.1 is a description of the system. In the second paragraph it is stated that "A manual selector switch allows the operator to equalize pump run times by periodically alternating the selection of the lead and backup pump." A sentence is being added before this sentence which states "The pumps are automatically sequenced to alternate starting in the normal mode of operation."

Evaluation: Adding this sentence does not cause an unreviewed safety question. This addition does not affect the licensing or design bases nor does it affect the operation or safety function of any system, structure, or component. It has no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. UFSAR Section 7.6.7.1 will be revised.

47 **Type:** UFSAR Change **Unit:** 0

Title: Change to UFSAR Section 7.7.1.2.1 "Full Length Rod Control System"

Description: UFSAR Section 7.7.1.2.1 "Full Length Rod Control System", Item 4 states : "Overlap between successive control banks is adjustable between 0 to 50 percent (0 and 115 steps) with an accuracy of +/- 1 step. This sentence is being changed to read "Overlap between successive control banks is adjustable between 0 to 50 percent (0 and 116 steps) with an accuracy of +/- 1 step. Overlap limits between successive control banks of 116 steps is specified in Section 2.4 of the Core Operating Limits Report (CNEI-0400-25 and CNEI-0400-25) and in Section 3.2 of the Reactor Operating Data" (ROD) book. This UFSAR correction will allow the UFSAR to match existing plant documents.

Evaluation: Revising the UFSAR to provide correct information that is already given correctly in plant documents does not constitute an Unreviewed Safety Question. No Technical Specification changes are required. UFSAR Section 7.7.1.2.1 will be revised.

48 Type: UFSAR Change **Unit:** 0

Title: Change to UFSAR Section 7.7.1.2.2 "Rod Control System Failures"

Description: UFSAR Section 7.7.1.2.2 "Rod Control System Failures" provides a discussion of the credible rod control equipment malfunctions which could potentially cause inadvertent positive reactivity insertions due to inadvertent rod withdrawal, incorrect overlap, or incorrect positioning of the rods. UFSAR Section 7.7.1.2.2 "Rod Control System Failures" is being revised to make a change to the text under the "Failures in the Overlap and Bank Sequence Program Control" paragraph. One item in this section states that: "Activation of the alarm light (urgent failure) on the cabinet front panel." is one of the actions that follow an urgent alarm. This sentence is not entirely accurate. These items are part of item two, "Failures in the Overlap and Bank Sequence Program Control" This item deals with both the logic cabinet and the power supply cabinet systems. Per the vendor manuals the alarm lightrs are generated for the power supply and/or the logic cabinet, depending on the source of the alarm. This item will be changed to state "Activation of the alarm light on the affected cabinet front panel".

Evaluation: Revising the UFSAR to provide correct information that is already given correctly in plant documents does not constitute an Unreviewed Safety Question. This UFSAR change will have no effect on the probability or consequences of accidents evaluated in the UFSAR. No Technical Specification changes are required. UFSAR Section 7.7.1.2.2 will be revised.

83 **Type:** UFSAR Change **Unit:** 0

Title: Change to UFSAR Section 9.2.5.3.1

Description: This revision to the UFSAR changes the description of the results of calculations performed by Duke Power Company. These calculations confirm the ability of the Standby Nuclear Service Water Pond (SNSWP) to provide cooling water to the plant for up to 30 days in the event of simultaneous Loss of Coolant Accident (LOCA) on one unit, Loss of Offsite Power (LOOP) on both units, Loss of Lake Wylie, and shutdown of the non-LOCA unit. The calculations have been refined to more accurately reflect the performance of the Nuclear Service Water System and the SNSWP. Specific changes to the calculation include a more detailed listing of the heat inputs to the pond, corrections to the analysis of the Nuclear Service Water System to more accurately reflect the flow capacity of the system, changes to the most severe meteorological conditions calculation, and inclusion of results of the Duke Power Gothic Program for calculation of the decay heat from the reactor core on the LOCA unit. Previously the decay heat input to the calculation was taken from curves supplied by Westinghouse. Duke has taken responsibility for these calculations and perforrns them in-house. In all cases, the results of these calculations provided more conservative results. The heat input to the SNSWP was increased as a result of these calculations and the cooling water flow rate was decreased. The results showed that the SNSWP is adequately sized to support the plant under these most severe conditions.

Evaluation: The UFSAR changes that are covered under this evaluation have no effect on the operation, design bases, or function of any structure, system or component. These changes do not increase the probability or the consequences of any accidents previously evaluated or the possibility of an accident of a different type than previously evaluated in the UFSAR. These changes do not increase the probability or consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR or the possibility of a malfunction of a different type than previously evaluated. These changes do not involve any safety or licensing issues, and do not revise any regulatory commitments. No Technical Specification changes are required. UFSAR Section 9.2.5.3.1 will be revised.

84 Type: UFSAR Change **Unit:** 0

Title: Change to UFSAR Section 9.5.7.2.2

Description: UFSAR Section 9.5.7.2.2, "Component Descriptions, " is being revised to delete extraneous detail related to the Diesel Generator Lube Oil System Lube Oil Heat Exchanger performance and reference appropriate Heat Exchanger Specification Sheets. The need to revise the applicable UFSAR section arose during review of related documents for Minor Modifications to correct discrepancies first found on Lube Oil Heat Exchanger Specification Sheet . These discrepancies include:

1 . Heat Transfer Surface Area

2. Overall Heat Transfer Coefficient, "U"

3. Conflicting Tube Material specifications

UFSAR Section 9.5.7.2.2, Component Descriptions, implies the function of the Lube Oil Heat Exchanger is to cool the Emergency Diesel Generator Engine lubricating oil during engine operation to within specified engine operating temperature parameters. Additionally, UFSAR Sec 9.5.7.2.3, Instrumentation and Alarms, notes each diesel generator engine has a monitoring system equipped with both temperature and pressure alarm and trip switches to aid operators in diagnosing abnormal operating conditions. UFSAR Table 9-43 lists the Diesel Engine Alarm and Shutdown set-points related to the Lube Oil Heat Exchanger inlet and outlet temperatures.

The subject of this proposed UFSAR change, concerns data specified in UFSAR Section 9.5.7.2.2 related to Lube Oil Heat Exchanger performance. The basic revision is to delete the specified Heat Exchanger performance technical details and more appropriately and generally reference the Heat Exchanger Specification Sheet. Deletion of the data in the UFSAR will not change the function of the Lube Oil Heat Exchanger as outlined in the UFSAR referenced sections above. The proposed changes to the UFSAR will more appropriately align the basis for the Lube Oil Heat Exchanger performance to the designed heat transfer condition as provided on Heat Exchanger manufacturer's specification sheet to limit numbers of other references with duplicate or varying technical details.

Since the Lube Oil Heat Exchanger is cooled by the Diesel Generator/Engine Jacket Cooling Water System, a review was made of that system to determine if any related UFSAR basis calculations or descriptions were affected. As a result, calculation CNC-1223.59-01-0005, "Nuclear Service Water Flow and Fouling Acceptance Criteria on Diesel Generator/Engine Jacket Cooling Water System Heat Exchangers" was found that appears to be the basis of UFSAR Section 9.5.7.2.2. The Lube Oil Heat Exchanger data specified in UFSAR Section 9.5.7.2.2 appears to have been extracted from above calculation, which was primarily evaluating an optional Diesel Generator/Engine Jacket Cooling Water System Heat Exchanger design that incorporates the Lube Oil Cooler heat transfer as input.

The above calculation made generalized estimates of Lube Oil Heat Exchanger and

optional Diesel Generator/Engine Jacket Cooling Water System Heat Exchanger performance. The more appropriate reference for the UFSAR should be the manufacturer's specification sheet supplied with the Heat Exchanger package. The above calculation will not be affected by this UFSAR change.

The Lube Oil and optional Diesel Generator/Engine Jacket Cooling Water System Heat Exchanger were both manufactured and designed by Thermxchanger, Inc; and therefore, the heat transfer performance of the Lube Oil Heat Exchanger would have been taken into account in related Diesel Generator/Engine Jacket Cooling Water System Heat Exchanger design.

Since the Lube Oil Heat Exchanger function is specifically related only to the Diesel Generator/Engine operation, there is no inter-related effect on other systems, structures or components.

Evaluation: There is no unreviewed safety question associated with the activity. The activity has no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. No UFSAR change is required.

85 **Type:** UFSAR Change **Unit:** 0

Title: Change to UFSAR Table 9-21

Description: UFSAR Table 9-21 (Chemical and Volume Control System Design Parameters) will be revised. The table currently lists 9,851 gallons of 4 % boric acid solution as being required to meet cold shutdown requirements shortly after full power operation. The value of 9,851 gallons is an old requirement based on the initial cores loaded at Catawba Nuclear Station. Since that time, the enrichment has been adjusted to allow for longer cycles between refuelings. For each new core, a new Core Operating Limits Report (COLR) must be issued. One of the requirements needed to complete this report, is the amount of boric acid needed to meet cold shutdown requirements shortly after full power operation. Since the amount of enrichment of each core may be slightly different and a new Core Operating Limits Report must be issued with each new core. The UFSAR reference to the amount of boric acid required to meet the shutdown requirements in Table 9-21 will be revised to reference the valid Core Operating Limits Report for the present core.

Evaluation: The amount of boric acid required to safely maintain the required shutdown margin (SDM) following full power operations has always been given in the COLR manual. The original value listed in the UFSAR was the value used for an earlier core. Since this value changes with new core designs, the UFSAR should now reference the COLR. The COLR is a controlled document. There are no unreviewed safety questions associated with this UFSAR Change. No Technical Specification changes are required. UFSAR Table 9-21 will be revised.

73 **Type:** UFSAR Change **Unit:** 0

Title: Creation of Selected Licensee Commitment (SLC) 16.13-4

Description: Selected Licensee Commitment (SLC) 16.13-4, Minimum Station Staffing Requirements is being created to combine staffing requirements presently contained in the Technical Specifications, 10CFR50.54(m), the Catawba Fire Protection Program, the Catawba Emergency Plan, and other administrative documents, into one SLC for ease of use by the station staff. No actual changes were made to any staffing requirements.

Evaluation: There is no unreviewed safety question associated with this Selected Licensee Commitment. No technical requirements of any documents were affected by the changes. The creation of this Selected Licensee Commitment has no effect on the probability or consequences of accidents described in the UFSAR. No Technical Specification changes are required. UFSAR Section 16.13-4 will be added.

222 **Type:** UFSAR Change **Unit:** 0

Title: Revision to Selected Licensee Commitment (SLC) 16.9-4, "Fire Detection Instrumentation"

Description: Selected Licensee Commitment (SLC) 16.9-6 is being revised to add two fire detectors to Table 16.9-3. The two detectors will be added to the "Minimum Channels Operable" column for Fire Detection Zone 72. This will change the total for this Zone from 23 to 25 detectors. The Operations Shift Manager's Office area in the Control Room is no longer a separate area. This portion of the Control Room has been renovated such that it is now an integral part of Fire Detection Zone 72. Therefore, the two detectors located in the office are considered a part of Fire Detection Zone 72. These detectors have always been included in a test procedure and the testing requirements have been kept current.

Evaluation: There is no unreviewed safety question associated with this UFSAR change. The change will have no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. Selected Licensee Commitment (SLC) 16.9-6 will be revised to add two fire detectors to Table 16.9-3.

37 **Type:** UFSAR Change **Unit:** 0

Title: Selected Licensee Commitment 16.13-2

Description: This change modifies Selected Licensee Commitment (SLC) 16.13-2 to allow Reactor Operators/Control Operators to become Qualified Reviewers. The inclusion of these personnel is consistent with the definition of "key site supervisory staff' outlined in Section 13.1.2.2.2 of the Catawba UFSAR. A clarification of which additional exempt personnel are eligible to become Qualified Reviewers is also included.

Evaluation: These changes are purely administrative. No Plant Systems, Structures, or Components are modified or caused to operate in a different manner from current operation. No seismic, environmental, materials, or reactivity effects are created on plant Systems, Structures, and Components as a result of these changes. No new failure modes or effects or new types of system/component interactions will be introduced upon any Systems, Structures, or Components as a result of these changes. System, Structure, or Component behavior during steady state and transient conditions will be unaffected by any of these SLC changes. Plant response to any external phenomenon, natural or man-made, will not be affected. No plant transients or accident analyses will require revision as a result of these changes. No actual testing requirements are being modified by these proposed changes; both the existing testing frequencies and testing specifications and acceptance criteria for all Systems, Structures, and Components are being maintained. No actual regulatory commitments are being eliminated or reduced by these changes.

There are no unreviewed safety questions associated with this change to Selected Licensee Commitment 16.13-2. This change will not increase the probability or consequences of any accident evaluated in the UFSAR. The changes are purely administrative in nature and do not alter any technical requirements. They will have no effect upon the manner in which the plant is operated or maintained. No Technical Specification changes are required. UFSAR Section 16.13-2 will be revised.

34 **Type:** UFSAR Change **Unit:** 0

Title: Selected Licensee Commitment 16.7-11

Description: Selected Licensee Commitment (SLC) 16.7-11 is being revised to clarify that Digital Rod Position Indication (DRPI) is not required operable as long as all shutdown and control rods are fully inserted. The revision will also address providing other methods of preventing rod motion once the reactor trip breakers have been opened. During several previous outages, time was lost on turbine warming because it was thought that the reactor trip breakers could not be closed with DRPI inoperable. This revision does not change the intent of the SLC, but will clarify the intent to help prevent misinterpretation in the future.

The existing SLC states "One Digital Rod Position Indicator (excluding demand position indication) shall be operable and capable of determining the control rod position within plus or minus 12 steps for each shutdown or control rod not fully inserted." To eliminate any confusion, the following sentence will be added to the Bases section: "As long as all shutdown and control rods are fully inserted, digital rod position indication is not required to be operable." In addition, the following note will be added to the Remedial Action section: "Once the reactor trip breakers have been opened, alternate methods may be used to ensure there is no possibility of rod motion. These methods are pulling fuses or sliding links in the rod control cabinets, tagging open the Motor Generator set feeder breakers from load centers LXC and LXD or tagging open the Motor Generator set output breakers. After one of these alternate methods is used, the reactor trip breakers may be reclosed if desired." The intent of the remedial action is to insert all shutdown and control rods and prevent any rod motion. This note will allow alternate methods of preventing rod motion once the reactor trip breakers have been opened. This allows the flexibility for closing the reactor trip breakers or energizing the Motor Generator sets, while still preventing rod motion.

Making this change to SLC 16.7-11 does not change the intent of the SLC. Inadvertent rod withdrawal will still be prevented.

Evaluation: This change to the SLC 16.7-11 will not increase the probability of accidents evaluated in the UFSAR. The initial action of opening the reactor trip breakers will still be done if all of the rods are not inserted. This change will allow other methods to be used once the initial action is completed. Allowing other methods to keep the rods inserted gives the plant the flexibility to close the reactor trip breakers if desired. There are no unreviewed safety questions associated with this change to the Selected Licensee Commitments. No Technical Specification changes are required. UFSAR Section 16.7-11 will be revised.

119 **Type:** UFSAR Change **Unit:** 0

Title: Selected Licensee Commitment 16.7.10 and 16.11-7

Description: The following changes are being made to the Selected Licensee Commitments (SLC):
1. Table 16.7- 10A, Table Notations, Item ***
2. Table 16.7-10A, ACTION C
3. Table 16.11-5, Item 5, Containment Air Release and Addition System
4. Addition of ACTION I to Table 16.11-5

Table 16.7-10A, Table Notations, Item ***

This item currently states: "When venting or purging from containment to the atmosphere, the trip setpoint shall not exceed the equivalent limits of SLC 16.11-18 in accordance with the methodology and parameters in the ODCM. When not..."

This item shall be revised as follows: "When venting or purging from containment to the atmosphere, the trip setpoint shall not exceed the equivalent limits of SLC 16.11-6 in accordance with the methodology and parameters in the ODCM. When not ... "

Table 16.7 -10A, ACTION C

This item currently states: With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge and exhaust valves are maintained closed.

This item shall be revised as follows: With less than the Minimum Channels OPERABLE requirement, operation may continue provided the Containment Purge and Exhaust Valves (VP) are maintained closed.

Table 16.11-5, Item 5, Containment Air Release and Addition System

This Table currently states:

Instrument	Min. Channels Operable	Applicability	Action
5. Containment Air Release and Addition System Noble Gas Activity Monitor - Providing Alarm (Low Range - EMF-39)	1	*	C

This Table shall be revised as follows:

Instrument	Min. Channels Operable	Applicability	Action
5. Containment Air Release and Addition System Noble Gas Activity Monitor - Providing Alarm (Low Range - EMF-39)	1	*	I

New Table 16.11-5, Action I

ACTION I -With the number of channels OPERABLE less than the Minimum Channels OPERABLE requirement, containment releases to the environment through this pathway may continue for up to 30 days provided that prior to initiating the release:

a. Vent system noble gas activity monitor providing alarm and automatic termination of release (Low Range - EMF 36) has at least one channel OPERABLE; or;
b. At least two independent samples of the containment atmosphere are analyzed, and at least two technically qualified members of the facility independently verify:

   1. The discharge valve lineup; and
   2. The manual portion of the computer input for the release rate calculations performed on the computer, or the entire release rate calculations if such calculations are performed manually.

Otherwise, suspend release of radioactive effluents via this pathway.

Evaluation: The first change corrects the cross-reference to SLC 16.11-6. This is an editorial correction and has no effect on unit operation. Therefore, this item will not be discussed any further.

The second item clarifies the applicability of SLC 16.7- 10 ACTION C to be limited to the Containment Purge and Exhaust Isolation Valves. Containment Purge and Exhaust System (VP) operation is automatically terminated by the containment radiation monitors (EMF 38, EMF39, EMF40) through the Sh Signal and by a Phase A Containment Isolation Signal. If the containment radiation monitors are inoperable then no provision exists (except for the Phase A Containment Isolation Signal) to automatically isolate the Containment Purge and Exhaust System if containment conditions change. The Containment Purge and Exhaust System typically operates with a nominal containment exhaust flow rate of approximately 25,000 cfm. As a result a change in containment conditions during a release could lead to a violation of the normal operation release limits (SLC 16.11-6) much more quickly than with the nominal 250 cfm flow rate of the Containment Air Release and Addition System. Therefore, it is appropriate for these valves to be secured closed in the event the containment radiation monitors are inoperable. It should be noted that Technical Specification 3.6.3 requires the Containment Purge and Exhaust System valves to be secured closed anyway in Modes 1 through 4. The Containment Air Release and Addition System is automatically isolated by the containment radiation monitors, the unit vent radiation monitors (EMF 35, EMF 36, EMF 37), and a Phase A Containment Isolation Signal. As a result, even if the containment radiation monitors are INOPERABLE, . Containment Air Release and Addition System would still be automatically isolated by the unit vent radiation monitors.

The third and fourth item extends the time the unit may operate with the containment radiation monitors INOPERABLE from 14 days to 30 days. The Selected Licensee Commitment requires either the unit vent radiation monitors (EMF 36) to be OPERABLE or independent analysis of redundant grab samples of the containment atmosphere to confirm that the release would not violate permissible environmental release limits in the event the containment radiation monitor (EMF 39) is inoperable. The 30 days is

consistent with the allowed out of service time for EMF 38 and 39 specified in Technical Specification 3.4.15, Reactor Coolant System Leakage Detection Instrumentation. The containment radiation monitors do not perform an Engineered Safeguards Feature Function and therefore the increased allowable out of service time does not affect any accident analysis in the UFSAR. Operation with the unit vent and containment radiation monitors out of service does increase the risk of a change in containment conditions not being detected while a containment release is in progress. If this occurred it is possible that the normal operation release limits (SLC 16.11-6) for the site could be challenged. The accident release limits of 10CFR 100 and GDC 19 are not affected by this change because as stated before the radiation monitors do not perform and Engineered Safeguards Function.

There is no effect on Technical Specifications 3.3.6, 3.6.3, and 5.5.5.

None of the systems affected by this change are accident initiators as described in the UFSAR. This change also does not change how these systems are operated. This change does not change operation of any of the leakage detection systems as specified in Technical Specification 3.4.15 and as a result does not affect the capability to detect a Rector Coolant System Leak. Therefore, the probability of an accident is not changed by this revision to the SLCs.

There are no unreviewed safety questions associated with this Selected Licensee Commitment change. No Technical Specification changes are required.

125 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change - Addition of Selected Licensee Commitment 16.6-5, "Residual Heat Removal/Containment Spray Sump Pump Interlock"

Description: A new Selected Licensee Commitment is being created. It will be Selected Licensee Commitment 16.6-5, "Residual Heat Removal/Containment Spray Sump Pump Interlock". The Selected Licensee Commitment establishes a requirement that the interlock shall be operable and a remedial action that when the commitment is not met, action shall be immediately initiated to implement alternate means of fulfilling the function of the interlock. The Selected Licensee Commitment also establishes an 18-month slave relay test frequency for the interlock relay. The purpose of the interlock is to ensure that the residual heat removal/containment spray sump pumps will trip on a safety injection signal on either unit and that they start on a sump high-high level signal. The interlock function will ensure that a passive ECCS failure outside containment, such as a pump seal failure, will not go undetected.

Evaluation: There are no Unreviewed Safety Questions associated with the creation of this new Selected Licensee Commitment. The manner in which the plant and its accident mitigating equipment are designed, operated, and maintained was not affected by these changes. No accident probabilities or consequences were affected by these changes. Similarly, no probabilities or consequences of equipment malfunctions were affected. The possibility was not created for any new type of accident or equipment malfunction. No safety margins were reduced by the proposed changes. No Technical Specification changes are required. Selected Licensee Commitment 16.6-5 will be added to the UFSAR.

231 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Figure 3-221

Description: The UFSAR will be changed to match the model presented in UFSAR Figure 3-221 to the actual calculation, CNS 1150.04-00-0006 where this model was developed. Figure 3-221 represents the model, mass weights and their location, and properties used for generating the seismic response spectra for Nuclear Service Water Pump House Structure. The model shown in the figure does not show all the nine mass points of the model used in the calculation. Also values of polar moment of inertia are not shown correctly in Figure 3-221 compared to values in the calculation. The correct values for polar moment of inertia are 1364 000 Ft^4 instead of 971 000 Ft^4 for members 1 through 6 and 742000Ft^4 instead of 543000 Ft^4 for members 7 and 8. This change is not a result of a new calculation but a correction of UFSAR Figure 3-221 to conform to the original calculation where the model and values shown in the figure were developed. Hence, this change does not affect the design and safety of Nuclear Service Water Pump Structure or attachment of electrical and mechanical equipment to it.

Evaluation: There are no Unreviewed Safty Questions associated with revising this UFSAR figure. No Technical Specification changes are required. UFSAR Figure 3-221 will be revised.

177 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Figure 3-282 and Figure 3-283

Description: A comparison of analyses of the Reactor Building operating floor by ELAS and STRUDLE computer programs was performed in engineering calculation CNC-1144.03-14-0003. The comparison results developed in the calculation were represented in the UFSAR as Figures 3-282 and 3-283. However, where results in calculation CNC-1144.03-14-003 were calculated and represented in "fraction of span", the results in Figures 3-282 and 3-283 were shown as "spans in feet". Therefore, representation of spans in the figures is incorrect. The figures will be changed to show "fraction of span" to match the analysis in the calculation.

Evaluation: There is no unreviewed safety question associated with changing the UFSAR figures to show "Fraction of Span" rather than "Feet". No Technical Specification changes are required. UFSAR Figures 3-282 and 3-283 will be revised.

178 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Figure 3-286

Description: Missing information is being added to UFSAR Figure 3-286. UFSAR Figure 3-286 lists criteria for the seismic design of block walls. These criteria were developed in engineering calculation CNC-1139.14-14-0001. In UFSAR Figure 3-286, several references are made to Table 3.8.4-1, but this Table is not included in the UFSAR. Also in the calculation determining the spacing of vertical steel members, how the value of W=3.25 was obtained is not presented. All of this missing information which was supposed to be contained in Table 3.8.4.1 and the determination of how the value of "W", was calculated is in engineering calculation CNC-1139.14-14-0001 which develops and establishes the criteria for seismic qualification of block walls. This missing information does not present any new parameters but can be determined from the stated criteria in UFSAR Figure 3-286. Therefore adding this missing information does not make any material change in the criteria. The change does not affect the design and safety of the seismically designed block wall.

Evaluation: There is no Unreviewed Safety Question associated with adding this information to UFSAR Figure 3-286. Adding this information will have no effect on the probability or consequences of accidents evaluated in the UFSAR. No Technical Specification changes are required. UFSAR Figure 3-286 will be revised.

245 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Figure 3-7

Description: UFSAR Figure 3-7 shows the locations of the postulated breaks in the Primary Reactor Coolant Loop. In Westinghouse Topical Report WCAP-8172-A, Westinghouse performed an analysis for the break locations shown in Figure II-D of the Topical Report. Some of the locations do not correspond between the Westinghouse Figure and Catawba UFSAR Figure 3-7. This change will correct UFSAR Figure 3-7 to match the break locations shown in the Westinghouse document.

Evaluation: There are no unreviewed safety questions associated with this UFSAR change. No Technical Specification changes are required. This change corrects the locations of the postulated breaks in the primary coolant loop as analyzed by Westinghouse in their Topical Report WCAP-8172-A. The change does not affect the design and safety of the Primary Coolant Loop. UFSAR Figure 3-7 will be revised.

18 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Figure 6-13 "Drain Piping Arrangement - Refueling Canal"

Description: UFSAR Figure 6-13 is being revised to add details to the figure about where the figure was developed from. The figure in the UFSAR was developed from Catawba approved drawings. This change identifies these drawings.

Evaluation: Adding this information will not affect the probability or consequences of accidents analyzed in the UFSAR. This addition will not affect the operation, design bases, or function of any plant system, structure, or component . There are no unreviewed safety questions associated with this UFSAR change. No Technical Specification changes are required. UFSAR Figure 6-13 will be revised.

101 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Figure 7-18 "Rod Control System Block Diagram"

Description: The Catawba UFSAR, 1998 Update, Figure 7-18 "Rod Control System Block Diagram" provides a simplified block diagram of the rod control system beginning with the rod control system inputs (process control, nuclear instrumentation, Solid State Protection System (SSPS), and manual actuation) through the rod control logic and power cabinet circuitry and ending with the control rod drive mechanism. The rod control system input from process control Loops 1 through 3 Tavg signal are shown as the only inputs to the summing amplifier Auctioneer Unit Highest Tavg. A review of plant documents has proven that this is not correct. Plant drawing CNM 1399.03-0023-032, Rev. 4, provides a block diagram of the inputs to the amplifier Auctioneer Unit Highest Tavg showing input from Loops 1 through 4 Tavg signals. Changing UFSAR Figure 7-18 to include a Tavg signal to the summing amplifier Auctioneer Unit Highest Tavg from Loop 4 does not change any assumptions or inputs to the safety analyses.

Currently Figure 7-18 The summing amplifier Auctioneer Unit Highest Tavg does not show an input from process control Loop-4 Tavg signal. Figure 7-18 will be changed by adding an input from process control Loop 4 Tavg signal.

Evaluation: UFSAR Figure 7-18 will be changed to reflect the rod control system input from process control Loops 1 through 4 Tavg signal, based on a review of CNM 1399.03-0023-032, Rev. 4, CNM 1399.03-0023-023, Rev. D4, CNM 1399.03-0124-039, Rev. D3, and CNM 1399.03-0023-010, Rev. 5. Correcting the figure does not constitute an unreviewed safety question. The correction does not change the licensing basis methodology. No changes to the Technical Specifications are required. UFSAR Figure 7-18 will be revised.

114 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Figure 7-21

Description: UFSAR Figure 7-21 provides a simplified block diagram of the pressurizer pressure control system beginning with input signals from the pressurizer pressure channels, a reference pressure signal, and the remote manual positioning signal (spray controller only) through the applicable controllers to the end device (power operated relief valves, heater control, and spray valves). Pressurizer pressure Channels 1 through 4 provide input to the Pressurizer Pressure Control System. The current UFSAR figure shows only a compensated pressure signal provided to PORV #2. A review of plant documents has proven that this is not correct. Plant drawings CNM 1399.03-0023-026, Rev. 3, (Process Control Block Diagram - Pressurizer Pressure Control) and CNM 1399.01-0024-011, Rev. 5, (Functional Diagrams - Pressurizer Pressure and Level Control) provide a correct, functional illustration of the Pressurizer Pressure Control System. Changing UFSAR Figure 7-21 to present an accurate description of the Pressurizer Pressure Control System does not change any assumptions or inputs to the safety analyses.

Based on a review of CNM 1399.03-0023-026, Rev. 3, and CNM 1399.01-0024-011, Rev. 5, UFSAR Figure 7-21 will be replaced with a figure which gives a better representation of the pressurizer pressure control system. This will be an extensive revision of the Figure.

Evaluation: There is no unreviewed safety question associated with this UFSAR revision. The revision has no effect on the probability or consequences of accidents analyzed in the UFSAR. No changes to the Technical Specifications are required. UFSAR Figure 7-21will be revised.

115 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Figure 7-22

Description: UFSAR Figure 7-22 provides a simplified block diagram of the pressurizer level control system beginning with input signals from the pressurizer water level channels, a reference temperature signal from auctioneered Tavg, charging flow signal, through the applicable controllers to the charging flow control valve position. Pressurizer Level Channels 1 through 3 provide input to the Pressurizer Level Control System to control level by varying the charging flow from the Chemical and Volume Control System Pumps. The current UFSAR figure shows only a signal to the charging flow control valve position. A review of plant documents has proven that this is not correct. Plant drawing CNM 1399.03-0023-027, Rev. D3A, (Process Control Block Diagram - Pressurizer Level Control) provides a technically correct, illustration of the Pressurizer Level Control System, with the exception of the Positive Displacement Pump. Replacing the figure with a modified version of drawing CNM 1399.03-0023-027, Rev. D3A, including a note stating the Centrifugal Charging Pumps provide charging flow and the Positive Displacement Pump is removed from service, will provide a better representation of the pressurizer level control system

UFSAR Figure 7-22 will be replaced with a modified version of Drawing CNM 1399.03-0023-027, Rev. D3A. This will be an extensive revision of the figure.

Evaluation: There is no unreviewed safety question associated with this UFSAR revision. The revision has no effect on the probability or consequences of accidents analyzed in the UFSAR. No changes to the Technical Specifications are required. UFSAR Figure 7-22 will be revised.

70 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Figure 7-24

Description: Nuclear Station Modifications CN-11168 and CN-20544 installed a Digital Feedwater Control System (DFCS) to control the water level in the Steam Generators. The existing Figure 7-24 in the FSAR does not accurately reflect the true workings of the system. Drawings CNM-1399.01-0024 (sheet 20) and CNM-2399.01-0002 (Sheet 20) were provided by Westinghouse to graphically explain the workings of the DFCS. These Westinghouse drawings were simplified and used as a replacement for the existing Figure 7-24.

Evaluation: There are no unreviewed safety questions associated with this UFSAR change. This change has no effect on the operation, design bases, or function of any structure, system or component. The change will have no effect on the probability or consequences of accidents evaluated in the UFSAR. No Technical Specification changes are required UFSAR Figure 7-24 will be revised.

116 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Figure 7-25

Description: UFSAR Figure 7-25 provides a block diagram of the steam dump control system beginning with the inputs from auctioneered Tavg, Reference Tavg, Treference from Turbine Pressure, Reactor Trip, Turbine Impulse Pressure, and manual control. The signals are processed and provide signals to trip or modulate open the steam dump valves. The figure number listed in the note at the bottom of the page is incorrect. The identifiers for the Tavg signal inputs to the summing amplifiers do not agree with drawing CNM 1399.01-0024-010, "Catawba Units 1 and 2, Functional Diagrams, Steam Dump Control." The diagram shows an output signal from the "Load Rejection Control or Plant Trip Control" block which modulates open the condenser dump valves. A review of drawing CNM 1399.01-0024-010 has identified that this is not correct. Plant drawing CNM 1399.01-0024-010 provides the correct terminology for the Tavg signal inputs to the summing amplifiers and identifies that all of the dump valves can be modulated. The referenced figure should be UFSAR Figure 7-2, sheet 10.

Figure 7-25 will be revised by changing the figure number listed in the note at the bottom of the page, changing the identifiers for the Tavg signal inputs to the summing amplifiers to agree with CNM 1399.01-0024-010, and by deleting "condenser" from "modulate condenser dump valves" at bottom of page.

Evaluation: There is no unreviewed safety question associated with this UFSAR revision. The revision has no effect on the probability or consequences of accidents analyzed in the UFSAR. No changes to the Technical Specifications are required. UFSAR Figure 7-25 will be revised.

117 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Figure 7-31

Description: UFSAR Figure 7-31 provides a simplified block diagram of the Rod Control System beginning with input signals from reactor control, a manual switch, and multiplexing circuits through the pulsers, cyclers, switches, and power cabinets to the end device (control and shutdown banks rod drive mechanisms). The current UFSAR Figure 7-31 does not show:
1) a signal from the reactor control system to the bank selector switch, 2) the in-hold-out switch, 3) the complete set of power cabinets, and much more. Vendor manual CNM 1399.05-0181-001, "Full Length Rod Control System, Volume 1 - Control System," Figure 1-2 "Sequencing of Groups Within a Bank" and Figure 1-4 "DCP/DDP Full Length Rod Control System Block Diagram provide a correct, functional illustration of the Rod Control System.

UFSAR Figure 7-31 will be revised by replacing it with a figure which is a combination of Figure 1-2 and Figure 1-4 from CNM 1399.05-0181-001, "Full Length Rod Control System, Volume I - Control System," revision date 11/02/1994.

Evaluation: There is no unreviewed safety question associated with this UFSAR revision. The revision has no effect on the probability or consequences of accidents analyzed in the UFSAR. No changes to the Technical Specifications are required. UFSAR Figure 7-31 will be revised.

124 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 10.4.8.2

Description: The UFSAR will be revised to change the means for isolating the Steam Generator Wet Lay-up Recirculation System from the Steam Generator Blowdown System from locked closed isolation valves to blind spectacle flanges. The UFSAR describes the Steam Generator Wet Lay-up Recirculation System as normally isolated from the Steam Generator Blowdown System by isolation valves that are locked closed. The UFSAR change will modify this description to portray the current alignment of the system which maintains the valves as normally open with blind spectacle flanges installed in the piping between the Steam Generator Wet Lay-up Recirculation System and the Steam Generator Blowdown Systems. The blind spectacle flanges have been installed previously and the subject valves were removed from the locked closed valve program under another modification. This change will be made to make the UFSAR consistent with the current plant configuration.

Evaluation: The Steam Generator Wet Lay-up Recirculation System is connected to the Steam Generator Blowdown System downstream of the containment isolation valves for each Steam Generator. The Steam Generator Wet Lay-up Recirculation System is physically isolated from the Steam Generator Blowdown System via the installation of blind spectacle flanges in the Steam Generator Wet Lay-up Recirculation System lines upstream of valves 1(2)BW6, 15, 22 and 31. The spectacle flanges are rotated to the open position, via the Work Order Process, every time the Steam Generators are placed in wet recirculation during outages. The spectacle flanges are rotated and the blind ends installed when the Steam Generators are returned to service.

Plant Engineering Procedure 3.18 "Locked and Administratively Controlled Valves" provides a listing of Specific Regulatory Requirements that specify locked closed valves. Each of these references was reviewed to determine if there were applicable criteria for maintaining the Steam Generator Wet Lay-up Recirculation System isolation valves in the locked closed position.

The applicable SER and UFSAR Sections were screened. The UFSAR sections that detail containment isolation functions, 3.1 and 6.2.4.2 were reviewed. Since the Steam Generator Wet Lay-up Recirculation System is connected to Steam Generator Blowdown System downstream of the containment isolation boundary, the locked closed designation for containment isolation criteria do not apply to the isolation valves in the Steam Generator Wet Lay-up Recirculation System. The subject valves in the Steam Generator Wet Lay-up Recirculation System are not listed in Table 6-77, which lists valves credited with containment isolation functions.

A review of the Selected Licensee Commitments failed to identify any commitments that reference these valves or their functions. A review of calculation references in Plant Engineering Procedure 3.18 also failed to identify any applicable references to the Steam Generator Wet Lay-up Recirculation System valves.

These Steam Generator Wet Lay-up Recirculation System valves were reviewed against the locked closed requirements delineated in Technical Specification Section 3.6.3. Per the requirements in this Section, the the Steam Generator Wet Lay-up Recirculation

System valves do not meet the criteria for inclusion into this section of the Technical Specifications. A review of the piping classification and associated pressure vessels failed to identify any application of the ASME code that would have required the valves to be locked closed.

There are no unreviewed safety questions associated with this UFSAR change. This change will have no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. UFSAR Section 10.4.8.2 will be revised.

228 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 3.7.1.2

Description: A missing figure is being added to the UFSAR for comparison of Case I and Case II described in Section 3.7.1.2 of the UFSAR. This comparison was done to establish the final system period intervals for the generation of time history response spectra. The missing figure was added to the Catawba PSAR as Figure 3.7.1-1 in response to question Q3.8 from the NRC.

Evaluation: There are no Unreviewed Safty Questions associated with adding this figure to the UFSAR. The missing figure does not represent any new design or change of design parameter. Therefore adding the missing figure does not create any design or safety concern. No Technical Specification changes are required. A new UFSAR figure will be added.

175 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 3.8.3.1.10

Description: UFSAR Section 3.8.3.1.10 paragraph 3 states that high strength bolts are used for Ice Condenser support columns. In fact these bolts are ASTM A-36 bolts according to plant drawings. These bolts were qualified as A-36 bolts in calculation CNC-1144.04-01-0001. Therefore the statement about high strength bolting is incorrect. The UFSAR text will be revised to show the correct type of bolting material

Evaluation: There is no unreviewed safety question associated with changing the bolting material description in UFSAR Section 3.8.3.1.10. The existing bolting material is qualified as A-36 bolting as shown on design drawings. No reanalysis of these bolts is required. No Technical Specification changes are required. UFSAR Section 3.8.3.1.10 will be revised.

176 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 3.8.5.4

Description: UFSAR Section 3.8.5.4 states that allowable bearing capacity of the rock formation of the Reactor Building is 965000psf. This value is not correct but is the ultimate bearing capacity of the rock foundation. The correct value for thie allowable bearing capacity is 100000 psf or more as per Section 6.1.2 of Engineering Calculation CNC-1110.00-00-0001. In analysis of the Reactor Building foundation the allowable bearing capacity of 100000 psf was used.

Evaluation: There is no unreviewed safety question associated with changing this item of information in UFSAR. The correct value was used in the analysis of the Reactor Building foundation. No Technical Specification changes are required. UFSAR Section 3.8.5.4 will be revised.

109 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 5.4.1.3.1

Description: In UFSAR Section 5.4.1.3. 1, the third paragraph states "...which also assures that adequate core cooling is provided to permit an orderly reduction in power if flow from a reactor coolant pump is lost during operation." This paragraph should be changed to "The Reactor Trip System ensures that pump operation is within the assumptions used for loss of coolant flow analysis." This change corrects the paragraph to agree Duke Power Catawba Nuclear Station Technical Specifications. The issue of allowing an orderly reduction in power below 48% is not currently supported by station Abnormal Procedures (AP).

Evaluation: This change does not involve an unreviewed safety question. The change has no effect on the probability or consequences of accidents evaluated in the UFSAR. No Technical Specification changes are required. UFSAR Section 5.4.1.3.1 will be revised.

110 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 5.4.1.3.1

Description: UFSAR Section 5.4.1.3.1 Design Evaluation; Pump Performance, states that the Reactor Coolant Pump curve has a "knee" at about 45 percent of design flow. The Reactor Coolant Pump curve included in the UFSAR was previously changed to utilize the correct pump curve. The new curve has a "knee" at about 62 percent of design flow. This change corrects the reference to the correct number for the new curve. This deviation in the trend of the pumps performance is a result of flow separation at the pumps diffuser and introduces no operational restrictions, since the pump normally operates at 100% flow.

Evaluation: This change does not involve an unreviewed safety question. The change has no effect on the probability or consequences of accidents evaluated in the UFSAR. No Technical Specification changes are required. UFSAR Section 5.4.1.3.1 will be revised.

127 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 6.2.5.7

Description: It has been determined that the Hydrogen Mitigation System at Catawba can no longer rely on AC Delco 12G glow plugs due to recent failures experienced with these glow plugs. Testing was performed to identify a suitable replacement. Champion model CH78 glow plugs were selected as the acceptable alternative to the AC Delco 12G plug. This modification will update the applicable Bills of Material and the affected drawings. The UFSAR is to be changed as follows:

Present Wording: "The HIS utilizes glow plugs manufactured by General Motors (GM) AC Division. The igniter is powered directly from a 120/13V ac transformer."

Revised Wording: "The HIS utilizes automotive diesel engine glow plugs. The igniter is powered directly from a multi-tap 120/10,12,14,16,18 VAC transformer.

General Motors never actually manufactured the previously used AC Delco 12G glow plugs. The manufacturer of the 12G and 7G glow plugs was Delphi. The present manufacturer of the AC Delco 12G glow plug is Wellman Thermal Systems. Since Delphi is no longer producing these glow plugs and the plugs produced by Wellman are unsatisfactory for our application, reference to the manufacturer and AC Delco is inappropriate.

Likewise the glow plug transformer was never at 120/13 Vac. A multi-tap transformer has always powered the glow plugs. The plugs were found to perform satisfactorily when powered from the 12-volt tap and had previously been powered from the 14-volt tap. Reference to the specific tap setting is inappropriate as the critical issue is maintaining adequate temperature at degraded bus voltage conditions in order to satisfy the hydrogen ignition requirements.

Current Wording: "The number of igniters on each circuit can range from 1 to 10."

Revised Wording: "The number of igniters on each circuit can range from 1 to 6."

This change is necessary because as presently designed, the maximum number of igniters per circuit is 6.

Additionally two typographical errors were corrected.

Evaluation: This activity is the revision of the Bills of Material for the Hydrogen Mitigation System such that known defective AC Delco 12G glow plugs will no longer be purchased and installed in the plant. Testing has been performed in order to identify glow plugs that are capable of satisfying the design basis requirements for post-accident ignition of hydrogen gas. Champion CH78 glow plugs were demonstrated by testing to satisfy these requirements. Original testing prior to installation of the Hydrogen Mitigation System identified AC Delco glow plugs as being the acceptable component. However over the years, the original model 7G glow plug became obsolete and was replaced by the AC Delco 12G glow plug. Later, the production of the AC Delco glow plug by Delphi ceased and Wellman Thermal Systems started production of the AC Delco 12G. Wellman elected

to modify the internals of the AC Delco glow plug they produce in order to improve automotive diesel engine performance. This modification has rendered the AC Delco 12G plug manufactured by Wellman unable to meet the design requirements for post-accident hydrogen mitigation.

This UFSAR change replaces one style of glow plug with another. The Hydrogen Ignition System is not an accident initiator. Testing has been performed on the replacement glow plugs in order to ensure that the replacement plugs are capable of satisfying the design basis requirements for the system. The replacement plugs satisfactorily completed this testing. Therefore the proposed activity does not increase the probability of occurrence of an accident previously evaluated in the UFSAR.

The replacement of AC Delco 12G glow plugs with Champion CH78 glow plugs will enable the Hydrogen Mitigation System to continue to satisfy its design basis requirements. The minor modifications to the Bills of Material are necessary to ensure that the glow plugs capable of satisfying the design bases are installed in the plant in place of the listed AC Delco plugs which are known to be deficient. The revision to the UFSAR Sections is necessary as these sections contained outdated information. There are no Unreviewed Safety Questions associated with this UFSAR change. No Technical Specification changes are required. UFSAR Section 6.2.5.7 will be revised.

49 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 6.3.2.2

Description: UFSAR Section 6.3.2.2 currently states that tilting disc check valves are used in applications of four inch and larger. This is incorrect. Swing check valves are actually installed in the plant in these applications. Westinghouse Pressurized Water Reactor designs, including Catawba, typically utilize swing checks for greater than four inch ECCS applications and this is considered the industry standard. This change will eliminate the reference to tilting disc check valves for ECCS applications four inch and larger.

The ECCS check valves originally installed when the plant was constructed were swing check valves, and they have never been replaced. All the testing associated with these check valves, including the ASME IWV program, is designed for swing-type valves.

Evaluation: No physical changes will be made to the plant as a result of this change to the UFSAR. The change in the description of the valves in Section 6.3.2.2 will not affect the probability of occurrence of any accident evaluated in the UFSAR. There is no unreviewed safety question associated with this UFSAR change. No Technical Specification changes are required. UFSAR Section 6.3.2.2 will be revised.

88 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 6.3.3 and 15.6.3

Description: UFSAR Sections 6.3.3 and 15.6.3 are being revised in order to describe the behavior of the Steam Generator Leakage Monitors in response to a Steam Generator Tube Rupture Event. Text will be added to state that the steam generator leakage monitor adjacent to the affected main steam line will alarm based on increased Nitrogen-16 activity. This is only a text change to the UFSAR and no modification to the physical plant is being performed.

Evaluation: This is only a text change to the UFSAR . No modification to the physical plant is being performed. There are no unreviewed safety questions associated with this UFSAR change. No Technical Specification changes are required. UFSAR Sections 6.3.3 and 15.6.3 will be revised.

145 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 6.3.5.4

Description: The alarms for the Refueling Water Storage Tank Level are based on any one channel instead of the 2 out of 4 channels stated in this section of the UFSAR. The Refueling Water Storage Tank Lo Level setpoint also de-energizes the Refueling Water Storage Tank heaters. During outages, Mode 5, 6 and No Mode, it is desirable to maintain heating of the Refueling Water Storage Tank even below the norrnal Lo Level setpoint. This was added for clarification only.

Evaluation: There is no Unreviewed Safety Question associated with this UFSAR change. Correction of inaccurate UFSAR information does not result in an Unreviewed Safety Question. This UFSAR Change has no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. UFSAR Section 6.3.5.4 will be revised.

122 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 6.7.15.2 and Figure 6-180

Description: The text description in UFSAR Section 6.7.15.2 for the instrumentation associated with Equipment Access Doors of the Ice Condenser System is too detailed, does not discuss all access doors, and in some cases is incorrect. UFSAR Figure 6-180, which is referenced in UFSAR Section 6.7.15.2 , is incomplete and does not correctly reflect the wiring configuration for this subsystem. This figure does not provide any added benefit for describing the affected instrumentation, therefore it will be deleted.

The text description in the referenced section speaks of the hardware details of the subsystem. There is no basis for this detail being referenced in the UFSAR. This section should speak of the functional operation of the subsystem. This evaluation addresses the necessary changes to revise this section of the UFSAR.

Evaluation: The Ice Condenser System is an Engineered Safety Feature system designed to limit containment temperature, pressure, and iodine levels following certain design bases events. The Ice Condenser is provided with equipment and personnel access doors to access portions of the Ice Condenser. Instrumentation is provided to monitor the status of each door.

For access to the upper Ice Condenser region, two Equipment Access Doors are provided. Within each Equipment Access Door is a smaller Equipment Access Personnel Door. A single set of status indicating lights is available on a panel in containment to display the status of these doors, collectively. The "closed" status is displayed if all of these doors are closed and the Equipment Access Door Seals are inflated. If the seals are deflated, or any one of these doors is open, the "open" status light energizes.

For access to the Lower Ice Condenser region, a single Personnel Access Door is provided. Separate status indication for this door is provided on the same panel as for the Equipment Access Doors. The closed indication is displayed when the door is closed and latched. If the door is open, the "open" light energizes. A separate "Door Open" light located on the door also energizes.

A Control Room annunciator energizes if any of the above mentioned doors open, or if the Equipment Access Door seals deflate. In the closed position, the Equipment Access Doors provide a thermal, vapor, and pressure barrier between the Ice Condenser and Upper Containment. This instrumentation only serves to provide indication of the door status and, as such, is not related to nuclear safety and is not necessary to mitigate the consequences of a design basis accident. Additional instrumentation is provided to monitor the status of ice bed temperature and of the Lower Inlet Doors, which open during a design basis accident.

There are no unreviewed safety questions associated with this activity. This change will have no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. UFSAR Section 6.7.1.5.2 will be revised and UFSAR Table 6-180 will be deleted.

67 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 6.7.15.2, Figure 6-178, and Figure 6-179

Description: Discrepancies were noted in the text description of the Lower Inlet Door Position Indication section of UFSAR Section 6.7.15.2 for the Ice Condenser System. Discrepancies were also noted with UFSAR Figure 6-178 and Figure 6-179, which are referenced by the above mentioned sections. UFSAR Section 6.7.15.2 discrepancies are as follows:

Certain limit switches are mentioned as being connected in-series to the "Door Open" light. These switches are actually connected to the "Door Closed" light. References to Figure 6-178 should be to Figure 6-179. References to Figure 6-179 should be to Figure 6-178.

UFSAR Figure 6-178 discrepancies are as follows:

The figure title is incorrect.

A note on the figure references the Personnel Access Door, but this door is not shown on the drawing.

UFSAR Figure 6-179 discrepancies are as follows:

The figure title is incorrect.

A line is missing at the bottom of the drawing to connect the "Doors Closed" light with the circuit.

Evaluation: The Ice Condenser System is an Engineered Safety Feature designed to limit containment temperature, pressure, and iodine levels following certain design bases events. The Lower Inlet Doors of the Ice Condenser remain closed during normal operation to form a thermal barrier between the Ice Condenser and the lower containment atmosphere. During certain design basis events, the doors open to allow air and steam to enter the Ice Condenser.

Instrumentation is provided on each Lower Inlet Door to alert the appropriate station personnel if any door opens. Each door includes two limit switches for indication and alarm purposes. One limit switch operates zone monitor lights on a local panel. Lights are provided on this panel to indicate if all zone doors are closed or if any zone door opens. A second limit switch provides a Control Room alarm if any door opens. As part of the Inlet Door Position Monitoring System, this instrumentation must be operable during Modes 1 through 4 to ensure the capability is available to monitor the status of the doors. This instrumentation is not related to nuclear safety and is not necessary to mitigate the consequences of a design basis accident.

Regulatory Guide 1.70 provides the standard format and content of Safety Analysis Report information.

The only Ice Condenser information included in this Regulatory Guide which is related to the Lower Inlet Doors is the door surface area and the door opening pressure. The Inlet

Door Position Monitoring System is not listed.

This UFSAR change is necessary to reflect the actual field configuration of the Ice Condenser System Lower Inlet Door Position Indication System. This instrumentation is for monitoring purposes only. The accidents evaluated in the UFSAR are not affected by this change. Therefore this change will have no effect on the probability or consequences of accidents evaluated in the UFSAR. There are no unreviewed safety questions associated with this UFSAR change. No Technical Specification changes are required. UFSAR Section 6.7.15.2 and UFSAR Figure 6-178 and Figure 6-179 will be revised.

215 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 6.7.3.2

Description: Information in the UFSAR about ice condenser lattice frames will be corrected. During design basis accidents and events, the lattice frames help position ice baskets within the ice condenser and allow the passage of steam and air around the ice baskets. During normal operation the lattice frames permit basket installation and removal. Currently, the UFSAR states that the first lattice frame level is 15 feet above the ice condenser floor, and that the next seven levels are vertically spaced at 6 foot intervals. This UFSAR change states that the first lattice frame level is 15' 9" above the ice condenser floor, and that the next seven levels are vertically spaced at approximately 6 foot intervals. The change will bring the UFSAR into agreement with vendor documents which show that the first lattice frame level is 15' 9" above the ice condenser floor, and that the vertical spacing between the next seven lattice frame levels vary from 5' 11 5/8" to 5'9".

Evaluation: There is no unreviewed safety question associated with this UFSAR change. These changes have no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. UFSAR Sections 6.7.3.2 will be revised.

104 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 7.1.2.4.2

Description: Section 7.1.2.4.2 of the Updated Final Safety Analysis Report (UFSAR) provides a discussion of the design criteria that are considered in the design of the safety-related instrumentation and controls.

A Regulatory Guide (RG) review for the Catawba Nuclear Station identified that RG 1.29, Rev. 3 was adopted with comment and that the site needed to review UFSAR Section 7.1.2.4.2 for potential revision. A subsequent comparison review by plant staff between revisions 2 and 3 of RG 1.29 determined that Section 7.1.2.4.2 conformed to revision 3 of RG 1.29. A subsequent review identified that Revision 3 of RG 1.29 is referenced in UFSAR Section 1.7, page 1-36 and in a letter from M.S. Tuckman (Duke Power) to NRC, dated 6/3/99, "Nuclear Quality Assurance Program, Amendment 25", page 17-5.

The correction to Chapter 7, Section 7.1.2.4.2, of the UFSAR is as follows:

Section 7.1.2.4.2 - Regulatory Guide 1.29; Change revision number from revision 2 to revision 3.

Evaluation: UFSAR Section 7.1.2.4.2 is being changed to identify Revision 3 of Regulatory Guide 1.29 as the correct revision reference to which the plant safety-related instrumentation design conforms. Correcting this text does not constitute an unreviewed safety question. This correction has no effect on the probability or consequences of accidents analyzed in the UFSAR. No changes to the Technical Specifications are required. A change is required for UFSAR Section 7.1.2.4.2.

92 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 7.6.23, Revision to subtopics for Bypass Description

Description: Several UFSAR Section 7.6.23 subtopics describing bypasses associated with the Boron Dilution Mitigation System (BDMS) are not clear and concise as related to the actual system design and operation.

The intent for "Indication of Bypass" as defined by IEEE 279-1971 is that when a Safety System is removed from service, for testing or other reasons, and is incapable of performing its safety function (bypassed) that this Condition be indicated in the control room.

The safety-related instrumentation and controls of the BDMS are train related and do not include protection channels as defined in IEEE 279-1971. The requirements as defined in Section I of that standard are written for protection systems; therefore, those requirements are not directly applicable to the BDMS.

This UFSAR revision provides a more precise description of the operation for the designed Defeat Functions of the BDMS.

Evaluation: There is no Unreviewed Safety Question associated with this UFSAR change. The change has no effect on the probability or consequences of accidents analyzed in the UFSAR. There are no actual plant changes associated with this revision. No Technical Specification changes are required. UFSAR Section 7.6.23 will be revised.

89 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 7.6.3.1 and Figure 7-13

Description: Four changes are being made to the UFSAR to revise the description and logic diagram of the annunciator alarm circuit for the Cold Leg Accumulator Isolation Valves to reflect the as-built condition, and make editorial changes for clarity. As currently written, the annunciator circuit is described as including valve position signals from both a valve motor operator limit switch and a stem mounted limit switch. The as-built circuit includes only a motor operator limit switch signal. The existing figure and description correspond to a generic vendor alarming scheme for these valves, and the original intent was to follow the suggested design. However, when Catawba circuits were actually designed and installed, it was more prudent to use a different arrangement. The stem mounted limit switch was used in an additional safety grade monitor light circuit instead of the non-safety annunciator circuit, and an additional safety grade monitor light circuit was included using a motor operator limit switch. The as-built design includes all the intent of the generic circuit design, but uses a slightly different arrangement to assure separation of the safety grade and non-safety grade components. The change in the circuit design occurred during plant construction. The change was considered minor in nature, and thus did not require an FSAR impact review per the document control program then in force.

The input to the annunciator circuit from the Cold Leg Accumulator Isolation Valve is through an optical isolator, energized by a valve motor operator limit switch contact. As a consequence, when the valve motor power source is de-energized, the annunciator circuit becomes non-functional. The valves are positioned and power removed during normal operation, and also during cold operations. This condition is accommodated with approved operating procedures for start-up, shut-down, and accumulator operation. Valve position verification is accomplished by using one of the monitor light circuits which has a different essential power source. Using switch contacts in separate safety-related circuits which are continuously active is a more reliable arrangement. Should an analyzed accident occur while the valve circuit breaker is closed and the valve active, a signal from the SSPS would automatically move the valve to its safety position.

An editorial change will revise the wording of the last sentence in a paragraph to be compatible with the first, and revise "motive" to "motor" to match contemporary usage.

The referenced figure has been revised to reflect the aforementioned changes in the text of UFSAR Section 7.6.3.1. While developing the figure, consideration was given to including logic for the Auxiliary Shutdown Panel (ASP). This logic was not included because the ASP is not part of the licensing basis, would require extensive additions to the UFSAR which would not add to the licensing basis discussion/descriptions, and the FSAR was approved with the current figures which do not show ASP logic.

Evaluation: This change to the UFSAR is to revise the text and a figure describing a non-safety alarm to more accurately reflect the as-built configuration. The described annunciator circuit is isolated from the control circuitry and has no capability to affect the position of the valve it is associated with. During normal operation, the annunciator circuit is unpowered, and thus can have no effect on any evaluated accident. When the valve is powered, safety signals act to position the valve to its safety position in the event of an accident. The monitor light circuits are also totally separate from all control circuitry, and cannot affect

the position of the valve. There are no unreviewed safety questions associated with this UFSAR change. No Technical Specification changes are required. UFSAR Section 7.6.3.1 and Figure 7-13 will be revised.

50 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 7.6.4.1

Description: UFSAR Section 7.6.4 presents the Containment Pressure Control System (CPCS) instrumentation and controls which sense accident situations and prevent excessive depressurization of the Containment through inadvertent or excessive operation of the engineered safety features. The Containment Pressure Control System also functions to allow operation of the Containment Spray System only when it is required for reducing containment pressure and inhibits operation when not required for containment protection.

Section 7.6.4.1 is being revised to change it to to read: "Electrical power to each train of CPCS is supplied by a safety-related power source. The CPCS permissives for the NS containment spray valves, VX air return damper, and H2 skimmer inlet valves are supplied from uninterruptible Safety-Related power sources. The CPCS permissives for the NS pumps and VX air return fan motors are supplied on the first sequencer load group of Safety-Related power."

NS is the Catawba System designation for the Containment Spray System. VX is the Catawba System designation for the Containment Air Return and Hydrogen Skimmer System.

Evaluation: There is no unreviewed safety question associated with this UFSAR change. The change corrects inaccurate technical information pertaining to the Containment Pressure Control System. The change has no effect on the operation, design bases, or function of the CPCS and related structures, systems or components. The change does not affect the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. UFSAR Section 7.6.4.1 will be revised.

144 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 7.6.5.1

Description: UFSAR Section 7.6.5.1 is being revised to accurately reflect the current terminology used at Catawba Nuclear Station. UFSAR Section 7.6.5.1 uses the phrase "low-low level", which is an old Westinghouse nomenclature. The current Catawba nomenclature for the level that initiates automatic swapover is "lo level". A tabulation of the old Westinghouse nomenclature versus the current Catawba nomenclature is shown below for clarification as given in Calculation CNC-1210.04-00-0079.

Westinghouse	Catawba	Function
High Level	High Level	Protect Against Overflow
Lo Level	Makeup Level	Makeup To Refueling Water Storage Tank Required
Lo-Lo Level	Lo Level	Automatic Residual Heat Removal Pump Swapover
Empty Alarm	Lo-Lo Level	Containment Spray Pumps Must Be Secured

Also, the alarms for the Refueling Water Storage Tank Level are based on any one channel instead of the 2 out of 4 channels stated in this section of the UFSAR. This change concurs with the design's safety analysis, the plant's Technical Specifications and plant procedures. This change is made to correlate the plant's design documents to agree with the plant's calculated and verified design basis in the Safety Analysis. A Minor Modification (CE-7690) revised the affected plant design documents, but did not make the required UFSAR changes. No plant equipment or plant operations are affected by this change.

Evaluation: There is no Unreviewed Safety Question associated with this UFSAR change. Correction of inaccurate UFSAR information does not result in an Unreviewed Safety Question. This UFSAR Change has no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. UFSAR Section 7.6.5.1 will be revised.

100 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 7.7.1

Description: The Catawba UFSAR, 1998 Update, Section 7.7.1, provides a discussion of Plant Control Systems functions. Item four in this section states that following a normal operational transient, the pressurizer pressure is restored to the design pressure +/- 35 psi. Upon review of plant documents, it was determined that this value was not correct. Calculation CNC 1552.08-00-0023 presents the uncertainties associated with reactor coolant system pressure and provides a recommendation to continue using the existing +/- 30 psi allowance. In Corrective Action Program Report serial number C-99-04556, plant engineering identified, through a review of plant operating data, that the reactor coolant pressure is being controlled at 2235 +/-3 psi which is well within the controller uncertainty of +/- 8psi. The controller uncertainty of +/- 8psi is a component of the design pressure uncertainty (allowance) of +/- 30 psi. The smaller design pressure allowance of +/- 30 psi is a more restrictive control band for the pressurizer pressure control and thus is more conservative than the present value of +/-35 psi listed in the UFSAR.

The correction to Chapter 7, Section 7.7. 1, of the UFSAR is as follows:

Section 7.7.1 - Item four, first sentence states; "Maintains or restores the pressurizer pressure to the design pressure +/- 35 psi (which is well within reactor trip and relief and safety valve actuation setpoint limits) following normal operational transients that induce pressure changes by control (manual or automatic) of heaters and spray in the pressurizer."

This sentence will be changed to state; "Maintains or restores the pressurizer pressure to the design pressure +/- 30 psi (which is well within reactor trip and relief and safety valve actuation setpoint limits) following normal operational transients that induce pressure changes by control (manual or automatic) of heaters and spray in the pressurizer."

Evaluation: Based on a review of calculation CNC 1552.08-00-0023 (Protection and Safeguards Instrumentation Uncertainties Calculation), and the problem evaluation by plant engineering, it was determined that the UFSAR text should be changed to reflect the system design pressure allowance of +/-30 psi. During plant operation, following normal operational transients, the Pressurizer Pressure Control System maintains or restores the pressurizer pressure to the design pressure. Changing the statement to provide the correct design pressure allowance of +/-30 psi, per CNC 1552.08-00-0023, is actually more conservative than the current value of +/- 35 psi, listed in the UFSAR and does not constitute an unreviewed safety question. The change to a smaller pressurizer pressure control design pressure uncertainty allowance for normal operational transients is more conservative and results in a more restrictive (tighter band) at which pressurizer pressure is controlled. The correction does not change the licensing basis methodology. No changes to the Technical Specifications are required. UFSAR Section 7.7.1 will be revised.

251 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 7.7.1.2.1

Description: UFSAR Section 7.7.1.2.1, "Full Length Rod Control Systems," paragraph three currently states: "The shutdown banks are always in the fully withdrawn position during normal operation, and are moved to this position at a constant speed by manual control prior to criticality. A reactor trip signal causes them to fall by gravity into the core. There are five shutdown banks."
The wording of the paragrapgh will be changed to state: "The shutdown banks are moved to the fully withdrawn position at a constant speed by manual control prior to criticality. During normal operations the shutdown banks are kept in the fully withdrawn position, except during the rod movement surveillance required by Technical Specifications. A reactor trip signal causes them to fall by gravity into the core. There are five shutdown banks.

Evaluation: This change corrects UFSAR Section 7.7.1.2.1 to agree with Technical Specification Surveillance Requirement 3.1.4.2 which is already approved. This is a non-technical change. There are no physical changes to the plant or to operating procedures. These changes do not affect the mechanical and thermal performance of the fuel or its reliability. Reactivity is not affected. This change does not adversely affect the fuel or reactivity control component reliability or performance or the operation of the plant. There is no effect on plant setpoints, safety limits, or design parameters. There is no adverse effect on the nuclear fuel, the reactor coolant system or containment. Therefore this change does not increase the likelihood or consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR. The margin of safety defined in the Technical Specifications is not reduced. There are no unreviewed safety questions associated with this UFSAR change. No Technical Specification changes are required. UFSAR Section 7.7.1.2.1 will be revised.

65 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 7.7.1.3.1

Description: The Catawba UFSAR, 1998 Update, Section 7.7.1.3.1 provides a description of the monitoring functions provided by the power range nuclear instrumentation . In the last paragraph it is stated that: "For periods during which the alarm on flux difference is inoperable, the axial flux difference is logged every one hour for the first 24 hours and every half hour later until the alarm is available." Upon review of plant Technical Specifications 3.2.3 and Procedure PT/1(2)/A/4600/009 "Loss of Operator Aid Computer," it was determined that this is not correct. Surveillance Requirement 3.2.3.1 of Technical Specification 3.2.3 requires each operable flux difference channel be read and logged within one hour after the loss of the alarm and every hour thereafter, until the alarm is restored.

The correction to Chapter 7, Section 7.7.1.3. 1, of the UFSAR is:

Section 7.7.1.3.1 - Last paragraph, change sentence; "For periods during which the alarm on flux difference is inoperable, the axial flux difference is logged every one hour for the first 24 hours and every half hour later until the alarm is available." Change the sentence to state; "When the alarm on flux difference becomes inoperable and in Mode 1 at 50% rated power or greater, the axial flux difference is logged within one hour and every hour thereafter, until the alarm is available."

Evaluation: The UFSAR text was changed to reflect the surveillance required by the plant Technical Specifications upon loss of the flux difference alarm. This was based on a review of Technical Specifications 3.2.3 and procedure PT/1(2)/A/4600/009 "Loss of Operator Aid Computer." Changing the statement to provide the correct surveillance requirements does not constitute an unreviewed safety question. The correction does not change the licensing basis methodology. No changes to the Technical Specifications are required. UFSAR section 7.7.1.3.1 will be revised.

107 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 7.7.1.3.5

Description: Section 7.7.1.3 of the UFSAR provides a discussion of the plant control signals used for monitoring and indication. Section 7.7.1.3.5 will be revised to read: "A rod bottom signal for the rods in the digital rod position system is used to operate a control relay, which generates the "RPI AT BOTTOM ROD DROP" alarm".

Evaluation: There is no unreviewed safety question associated with this UFSAR change. The change has no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. UFSAR Section 7.7.1.3.5 will be revised.

153 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 7.7.2

Description: UFSAR Section 7.7.2 provides discussion of the analysis of the plant control systems design for high reliability in any anticipated operational occurrences. A discussion is given of the review of IE Bulletin 79-27, "Loss of Non-Class 1E Instrumentation and Control Power System Bus During Power Operation, Actions 1 through 3," and how Catawba complies. A description of the automatic functions of the non-1E auxiliary control inverter includes the following: "Relative to non-Class 1E non-safety-related power supply inverters; and 3) The auxiliary control inverter static switch will automatically transfer the load to the alternate source whenever the "inverter output voltage reaches 75%." A review of plant documents has proven that this is not correct. In plant drawing CNM 1358.02-0013 (14,15,16) "Inverter Final Test Report," a functional test of the inverters static switch is performed. In this test, the low voltage limit to initiate transfer was set at "minimum." Upon discussing this with the vendor, the actual value for the "minimum" voltage value recorded on the test report could not be verified. CNM 1358.02-0008-001, "Instruction and Operating Manual; Inverter Model SV12300/"FSNB/TSMB," Section five provides instructions for adjustment of the static switch voltage sense circuit with a recommended transfer value of 70% of nominal output voltage. The static switch transfer voltage value is not documented in the Duke response to IE Bulletin 79-27 or in calculation CNC-1381.06-0021, "240/120 VAC Auxiliary Control Power System Inverter Sizing". The voltage value is not verified in any plant procedure. Considering that the static switch low voltage transfer limit is not identified in any calculation or commitment, it is being deleted from the UFSAR because it is considered "excess detail" included in the FSAR at the time of initial licensing. Changing the UFSAR text by deleting the specific static switch voltage transfer limit does not change any assumptions or inputs to the safety analyses.

In Section 7.7.2, paragraph two, the following text will be deleted

"3) The auxiliary control inverter static switch will automatically transfer the load to the alternate source whenever the inverter output voltage reaches 75%.".

This text will be replaced with the following: "The non-class IE inverters provide the normal source of power to the auxiliary power panelboards. Each inverter normally feeds its associated panelboard, but during inverter undervoltage or overcurrent conditions, or during an inverter failure, an automatic static switch transfers the affected panelboard to an alternate supply provided from the regulated power distribution centers."

Evaluation: The UFSAR text should be changed to describe the actual inverter/static switch interactions (i.e., static switch transfer from the inverter to alternate source) on undervoltage, overcurrent, or inverter failure, setpoint values need not be included. Changing the text to provide the correct functional description without including the static switch undervoltage transfer setpoint does not constitute an unreviewed safety question. No Technical Specification changes are required. UFSAR Section 7.7.2 will be revised.

141 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 8.3.2.1.2.1.2

Description: The Catawba UFSAR, 1998 Update, Section 8.3.2.1.2.1.2 provides a discussion of the vital battery sizing, construction, location and operation. The following text is included: "Each vital battery consists of 60 cells in individual transparent, non-combustible, shock absorbing containers with covers, racks, and accessories." A review of plant documents has shown that this is not entirely correct. The 125 VDC Vital Instrumentation and Control Battery manuals do not specify that the battery containers are non-combustible. Table 3 of the Fire Hazards Analysis (Appendix A. 2, Part B) section of the Plant Design Basis Specification for Fire Protection (CNS 1465.00-00-0006) documents that the vital battery cell containers are considered to be a combustible material in the Unit 1 and 2 Battery Rooms. The analysis concluded that under worst-case conditions where the fire suppression system does not function, a fire in a battery room could possibly cause a loss of redundant trains. However, the 3-hour rated fire barriers would contain the fire within the fire area, and shutdown capability would be available from the Standby Shutdown System.

UFSAR Section 8.3.2.1.2.1.2 will be changed to delete "Non-combustible" from the following sentence; "Each vital battery consists of 60 cells in individual transparent, non-combustible, shock absorbing containers with covers, racks, and accessories."

Evaluation: There are no unreviewed safety questions associated with this UFSAR change. There is no actual change to plant systems, structures, or components. No changes to the Technical Specifications are required. UFSAR Section 8.3.2.1.2.1.2 will be revised.

66 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 9.2.1.3

Description: UFSAR Section 9.2.1.3, Safety Evaluation, relevant to the Nuclear Service Water System, describes operator actions upon a safety injection signal. The description includes actions taken by the non-LOCA operator to check for the operation of the Nuclear Service Water System pumps. This action is performed by the LOCA unit operator as outlined in procedure EP/1(2)/A/5000/E-0, Reactor Trip or Safety Injection and this section will be revised accordingly. Additionally, this section states that "if a single Nuclear Service Water System Pump is not operating, the crossover is left open and unnecessary components on the non-LOCA unit are secured to assure sufficient flow to the remainder of the system". While this is true, this action occurs as series of evaluations and determinations by the operators utilizing operating and abnormal operating procedures as necessary. The wording of this statement could be interpreted to mean that direct procedural guidance exists in one place for this action. For this reason, the wording of this sentence will be changed to read, " If a single Nuclear Service Water System Pump is not operating, the crossover remains open and unnecessary components on the non-LOCA unit would be secured to assure sufficient flow to the remainder of the system".

Evaluation: The Nuclear Service Water System , including Lake Wylie and the Standby Nuclear Service Water Pond (SNSWP), is the ultimate heat sink for various nuclear safety related heat loads during normal operation and design basis events. The Nuclear Service Water System supports Emergency Core Heat Removal (ECCS) operation by providing cooling to the Component Cooling System via the Component Cooling Heat Exchangers and also to the Diesel Generators (D/Gs) via the D/G Engine Jacket Water Cooler System Heat Exchangers. Other nuclear safety related loads include the Containment Spray Heat Exchanger and the Control Room Chiller Condenser. The Nuclear Service Water System also provides assured makeup to the Component Cooling System, Spent Fuel Pool, Auxiliary Feedwater supply and the Containment Seal Water Injection System.

The Nuclear Service Water System shall be capable of mitigating the consequences of a design basis event on one unit concurrent with a loss of offsite power on both units, assuming a single failure. The most stringent design basis event for the Nuclear Service Water System is a Loss of Coolant Accident (LOCA) on one unit. The following assumptions are further postulated for the Nuclear Service Water System design basis LOCA:

Normal Shutdown of remaining unit from normal operation.
Loss of Non-Emergency A/C Power (loss of offsite power) affecting both units.
Prolonged Drought in hot weather (maximum supply temperature/minimum supply volume).
Loss of Lake Wylie.
Single Active Failure.

A separate crossover line is provided for each unit in the Auxiliary Building. These lines are normally open to provide cooling flow to components on the Nuclear Service Water System non-essential headers and to provide flexibility in system operation. The supply header crossover valves and the non-essential header isolation valves close on the LOCA

unit on the phase B isolation signal. This assures adequate flow for the Containment Spray Heat Exchangers during sump recirculation operation. The supply header crossover valves on the non-LOCA unit remain open so that cooling can be maintained on the non-LOCA non-essential header. With this supply crossover valve open, there could be flow between the redundant channels. The system is protected against single active failure by interlocks and operator action.

The actions of Pperations personnel with respect to the Nuclear Service Water System in response to an accident are described in this UFSAR Section. The Nuclear Service Water System is not an accident initiator, therefore it does not increase the probability of occurrence of an accident previously evaluated in the UFSAR.

The UFSAR currently takes credit for operator action to ensure sufficient Nuclear Service Water System flow during the design basis accident if a single Nuclear Service Water System Pump is not operating. This is necessary to show that the Nuclear Service Water System is protected from a single active failure. Verification of the Nuclear Service Water System Pumps running is assured regardless which operator performs this function. The response of operations personnel to a single Nuclear Service Water System Pump not running remains unchanged, only the wording has been changed to more accurately reflect the actions taken.

There are no unreviewed safety questions associated with this UFSAR change. No Technical Specification changes are required. UFSAR section 9.2.1.3 will be revised.

136 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 9.2.7.3, Table 6-98, Table 9-15 and Figure 9-62

Description: UFSAR Section 9.2.7.3, Table 6-98, Table 9-15 and Figure 9-62 are being revised to change the specified Refueling Water Storage Tank volume from 350,000 gallons to correlate with the existing analyzed design document requirements including those in the Technical Specifications. The plant's accident safety analyses utilize the value of 363,513 gallons given in the Technical Specifications as its basis value and envelops it with a maximum value analyzed of 386,381 gallons and a minimum value analyzed of 348,426 gallons. The existing UFSAR value (350,000 gallons) and the Technical Specification value (363,513 gallons) are both enveloped in the existing Safety Analysis. No plant equipment or plant operations are affected by this change.

Evaluation: There is no unreviewed safety question associated with this UFSAR change. Correcting inaccurate technical information or deleting, adding or changing the text for enhancement or clarification does not result in an unreviewed safety question. No changes to the Technical Specifications are required. UFSAR Section 9.2.7.3, Table 6-98, Table 9-15 and Figure 9-6 will be revised.

218 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 9.3.5.5.6

Description: UFSAR Section 9.3.5.5.6 which addresses conductivity and pH of the Recycle Evaporator condensate, is being revised. The Recycle Evaporator receives preheated, degassed borated water. The flow is regulated to maintain constant level within the shell. The evaporator concentrates are continuously recirculated. The major portion of the vapors leaving the evaporator flow through the absorption tower where it contacts the vapor rising from the evaporator shell. To ensure boric acid is not carried over in the distillate, conductivity is measured and displayed to the operator. High conductivity is alarmed on the Recycle Evaporator control panel.

This change corrects the statement that there is pH instrumentation for the Recycle Evaporator. Due to the extremely pure nature of the recycle evaporator condensate, a decision was made during design/construction phase of the plant not to install the pH equipment. This change corrects a statement that implies the pH instrumentation is available to agree with the UFSAR Figures 9-100, 9-101, 9-102, 9-103, 9-104, and 9-105, which shows no pH instrumentation.

Evaluation: There are no unreviewed safety questions associated with this UFSAR change. This correction does not involve any change to operation, design basis or function of any structure, system or component. There are no physical changes or procedure changes required due to this UFSAR change. The corrections or changes do not involve any changes to the operation, design basis or function of any structure, system or component (SSC). No changes to the Technical Specifications are required. UFSAR Section 9.3.5.5.6 will be revised.

223 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change To Section 9.4.3.2.3, Table 6-100 and Table 6-101

Description: The Catawba UFSAR is being revised to incorporate two changes: (1) an editorial change to UFSAR Section 9.4.3.2.3 and (2) UFSAR Tables 6-100 and 6-101. The editorial change to 9.4.3.2.3 is made to correct an error in the description of how the Auxiliary Building Ventilation System is realigned on a high Unit Vent radiation alarm. The tables in Chapter 6 are being revised to more clearly describe how Catawba conforms to the applicable sections of Regulatory Guides 1.95 (Rev. 1) and 1.78 (Rev. 0). .

Evaluation: There are no unreviewed safety questions identified with these UFSAR revisions. The changes are editorial and have no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required.UFSAR Section 9.4.3.2.3 and UFSAR Tables 6-100 and 6-101 will be revised.

106 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 9.5.4.2.3

Description: Nuclear Station Modification CN-10758 (Unit 1) and CN-20130 (Unit 2) upgraded the Diesel Fuel Oil Recirculation System and removed the fuel oil recirculation filter inlet and outlet pressure indicators. Flow diagrams CN-1609-3.0, Rev. 19 and CN-2609-3.0 Rev 13 of the Diesel Generator Fuel Oil System show that instruments 1(2)FDPG5030/5031, Fuel Oil Recirculating Filter inlet and outlet pressure indicators are no longer installed. The UFSAR will be revised to correct this inaccurate information. The revision has no effect on the operation, design bases, or function of the Diesel Generator Fuel Oil system and related structures, systems or components

The correction to Chapter 9, Section 9.5.4.2.3, of the UFSAR is as follows:

Revise the list of instrumentation and alarms by deleting, "Fuel oil recirculation filter - inlet and outlet pressure indication."

Evaluation: Changing the UFSAR text by deleting references to the fuel oil recirculating filter inlet and outlet indicators does not constitute an unreviewed safety question. The revision has no effect on the probability or consequences of accidents analyzed in the UFSAR. No changes to the Technical Specifications are required. UFSAR Section 9.5.4.2.3 will be revised.

69 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Sections 11.2, 11.5 and 11.6

Description: UFSAR Sections 11.2, 11.5 and 11.6 of Catawba UFSAR are being updated. In addition, UFSAR Figure 2-4 requires a change. There are inaccuracies both technical and non-technical in these sections regarding the current operation of Liquid Radwaste Recycle System, determination of effluent monitor setpoints and other areas. The majority of the inaccuracies involve bases for monitor response and setpoints. Overall the operation of the Liquid Radwaste Recycle System and related systems will continue such that the amounts of radioactive material in the liquid effluents are reduced to assure that doses to individuals beyond the site boundary are within the limits specified in 10CFR50, Appendix I. The station will continue to follow all applicable Selected Licensee Commitments and Regulatory Guide 1.21. The changes described in this package do not adversely affect the intended function, operation or reliability of the nuclear safety related Residual Heat Removal System or the Containment Spray System Sump Pumps.

Evaluation: There are no unreviewed safety questions associated with this UFSAR change. No Technical Specifications changes are required. Changes are required to UFSAR Sections 11.2, 11.5, and 11.6

198 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Sections 6.7.1.2, 6.7.1.3, and 6.7.6.2

Description: During normal operation, the ice condenser floor is kept cold by a chilled glycol solution running though cooling coils embedded in the concrete wear slab. The floor cooling coils do not function during design basis accidents. Currently, the UFSAR states that the floor cooling coils originate and return to the Air Handling Unit return header, the design pressure of the floor cooling coils is 100 psi, that they are tested to 200 psi, and that the bleed flow rate is controlled by a temperature control valve. The UFSAR is being changed to state that the floor cooling coils originate and return to the Air Handling Unit supply header, the design pressure is 300 psi, that they are tested to normal system pressure, and that the operators manually control the bleed flow rate based on temperature. These changes were made to the ice condenser per Nuclear Station Modifications CN-10632/00 and CN-20011/00. These inconsistencies were found by a UFSAR Accuracy Review.

This UFSAR change pertains to the floor cooling system in the ice condenser. Per Nuclear Station Modifications CN-10632/00 and CN-20011/00 the floor cooling system was changed so that it originated and ended on the Air Handling Unit supply header. These modifications also increased the design pressure to 315 psia, and provided a manual throttle valve to control flow. Design documents were changed, but the UFSAR text that describes the floor cooling system was not changed. The 10CFR50.59 evaluations performed for these modifications concluded that they did not create any unreviewed safety questions. These 10CFR50.59 evaluations concluded that the modifications did not "appear significant enough to require inclusion in the UFSAR". However this information should have been included in the UFSAR.

Evaluation: There are no unreviewed safety questions associated with these UFSAR changes. No changes are being made to plant systems structures or components. No Technical Specification changes are required. UFSAR Sections 6.7.1.2, 6.7.1.3, and 6.7.6.2 will be revised.

197 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Sections 6.7.6.2 , 6.7.6.3, and Table 6-121

Description: UFSAR Sections 6.7.6.2 and 6.7.6.3 and UFSAR Table 6-121 were revised to correct information about the Ice Condenser glycol circulation pumps. This will make the UFSAR consistent with design drawings. No changes are made to the design bases or function of the glycol circulation pumps. Changes were made to the operation of the glycol circulation pumps, but the glycol circulation pumps do not operate to mitigate design basis events or accidents.

During normal operation, the glycol circulation pumps circulate ethylene glycol solution to convey heat from the heat transfer surfaces in the ice condenser normally to the Ice Condenser System chillers. Currently, the UFSAR states that there are 6 pumps installed, but only 4 are required to function. Station operating experience has shown that all of the pumps must normally run to provide adequate cooling flow. The flow diagrams show three pumps normally aligned to each unit. Without all three pumps running, a unit would be in danger of violating the Technical Specification value for ice condenser temperature, particularly during the summer months. This UFSAR change notes that there are six glycol circulation pumps with three pumps aligned to each unit, and deletes all wording which infers that there are spare glycol circulation pumps.

Evaluation: There are no unreviewed safety questions associated with these UFSAR changes. These changes do not increase the probability or consequences of accidents evaluated in the UFSAR and no new accidents scenarios are created by these changes. No changes to the Technical Specifications are required. UFSAR Sections 6.7.6.2 and 6.7.6.3 and UFSAR Table 6-121 were revised.

51 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Sections 9.3.4.2.2.1, 9.3.4.2.2.2, 9.3.4.2.3.7, and 9.3.4.2.3.11.

Description: UFSAR Section 9.3.4.2.2.1 states "The concentration of lithium-7 in the Reactor Coolant System is maintained in the range specified for pH control (0.2 to 2.2 ppm)." This specification is no longer utilized for Pressurized Water Reactor (PWR) chemistry. EPRI Primary Water Chemistry Guidelines emphasize the use of a coordinated boron/lithium program. Controlling reactor coolant pH to a specified target has been found to inhibit crud deposition/transport on fuel rods. The target pH can vary from plant to plant for control of different symptoms. At Catawba, the target pH was chosen to reduce the possibility of axial offset. This could change as different problems are encountered in the future. A concentration band of 0.2 to 2.2 ppm was found to be too wide, thus allowing wide pH swings. The pH swings would cause local crud release for deposit on other surfaces. The Primary Water Chemistry Program in the Duke Power System Chemistry Manual mirrors the EPRI Guidelines. The 9.3.4.2.2.1 statement will be changed to read " The concentration of lithium-7 in the Reactor Coolant System is maintained in the range specified for pH control per the coordinated boron/lithium program."

UFSAR Section 9.3.4.2.2.2 states "Hydrazine is not employed at any time other than startup from the cold shutdown state". Hydrazine is used to scavenge oxygen during plant startup but it is used for other purposes also. Hydrazine is added to the Residual Heat Removal System during non outage periods to scavenge oxygen. It was found that this oxygen could cause iron/cobalt to precipitate in the Residual Heat Removal System when the system is "valved in" during plant shutdown. This would increase dose rates during outages. Removal of the residual oxygen in the Residual Heat Removal System has been found to mitigate this increase in dose rates. Another use of hydrazine would be to enhance reducing conditions during the acid reducing phase of crudbust at shutdown. If residual hydrogen became depleted during this phase, hydrazine could be added to maintain suitable reducing conditions during this critical phase of crudburst. Hydrazine reacts with oxygen more efficiently in the 180 degree F. area and decomposes rapidly at higher temperatures thus having no effect on system components. The statement in 9.3.4.2.2.2 adds no value and could be misinterpreted that hydrazine is used for other purposes. This statement will be deleted.

UFSAR Sections 9.3.4.2.2.2 and 9.3.4.2.3.7 state a specification range of 25-35 cc/kg for hydrogen in reactor coolant. This range is incorrect. The actual range is 25-50 cc/kg. The 25-35cc/kg was considered an "optimal" concentration range for operation in the past based on Rev. 1 of the EPRI Primary Water Chemistry Guidelines. This presented evidence from laboratory studies that dissolved hydrogen increased the susceptibility of some mill-annealed Alloy 600 steam generator tubing to stress corrosion cracking in high temperature water. The 25-50 cc/kg range was still stated as the full range. Further EPRI study concluded that dissolved hydrogen in the range of 25-50 cc/kg did not have a significant effect on PWSCC initiation times. All subsequent revisions since Rev.3 of the EPRI Primary Water Chemistry Guidelines state that the 25-50 cc/kg range can be applied to all plants. EPRI has also found that operation in the higher range may help to mitigate AO. The present administrative limit (35-50 cc/kg) is recommended in the Duke Power System, Chemistry Manual. The Design Basis Document (CNS-1554.NV-00-0001,Rev. 13, Section 31.1.1.2.2) has the correct range of 25-50 cc/kg. This range will be corrected

to 25-50 cc/kg for normal operating conditions. The reactor is allowed to go critical at >= 15 cc/kg (consistent with Mode 2) and can be reduced to 15 cc/kg 24 hours before shutdown without entering an action level. Operation between 15 and 25 cc/kg during normal plant operation would denote action level 1. EPRI PWR Primary Water Chemistry Guidelines, Rev 4, Vol 1, states that computations of production rates of oxidizing species by radiolysis suggests a dissolved hydrogen concentration of significantly less than 15cc/kg is sufficient to scavenge the oxidizing species under all operating conditions.

UFSAR Section 9.3.4.2.3.11 states "A lithium-form cation resin and hydroxyl-form anion resin are charged into the demineralizer". A cost study showed that it was more feasible to lithiate a hydrogen-form resin on-line than to purchase lithium form resin. This technique produces in-situ lithium-form resin and thus yields the same product. The Design Basis Document CNS-1554.NV-00-0001, Rev.13, Section 31.1.1.2.3) does not state what form of resin is to be used in the mixed-bed dernineralizer.
The statements "Hydrogen form cation resin can also be charged into the demineralizer instead of lithium-form resin. The hydrogen-form resin is then converted to the lithium form on-line" will be added. These statements would allow the use of either hydrogen-form or lithium-form resin to be charged into the dernineralizers depending on availability or operating philosophy.

The cation bed has to be valved in manually by Operations while the mixed bed can be valved in by the Operator At The Controls (OATC). Utilization of the standby mixed bed demineralizer for routine lithium control is thus more resource efficient and has less chance of human error. This use of the standby mixed bed is compatible with the Design Basis Document description in that it can be lithiated for use as an in-service bed if the in-service bed became depleted. In that case, the cation-bed would be used for back-up lithium removal. The statements "It can also be used for lithium control. If used for this purpose, the cation resin in the standby demineralizer would be charged in the hydrogen-form" will be added to Section 9.3.4.2.3.11. These statements would clarify the role of the standby mixed-bed dernineralizer for lithium control.

Evaluation: There are no Unreviewed Safety Questions associated with these UFSAR changes. The changes presented here reflect current PWR chemistry practice per the EPRI Primary Water Chemistry Guidelines, Rev.4. for lithium and hydrogen control. The utilization of a mixed bed demineralizer in the hydrogen form has a significant advantage over the use of the cation demineralizer in lithium control since it can be valved in directly from the Control Room. The option to use lithium-form resin was retained. The addition of hydrazine to the Residual Heat Removal System during non outage periods as an alternative use of hydrazine is becoming an industry standard. Its use for this purpose could represent a significant dose savings. No Technical Specification changes are required. UFSAR Sections 9.3.4.2.2.1, 9.3.4.2.2.2, 9.3.4.2.3.7, and 9.3.4.2.3.11 will be revised.

57 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment 16.7-3 to add the "Precipitation" loop as a required channel for the meteorlogical system

Description: This change will add the Precipitation Channel as a required meteorological variable to Selected Licensee Commitment (SLC) 16.7-3. The precipitation equipment/channel is in service and is currently providing input to the "RADDOSE-V" computer program. In addition, an administrative requirement is added to generate a Special Report to the NRC if the annual data recovery goal of greater than or equal to 90% is not met, as required by Reg. Guide 1.23 Revision 0.

Radiation Protection and Emergency Preparedness use the computer program RADDOSE-V to calculate the diffusion estimates of the plume following a release. This program uses precipitation as one if its inputs. The precipitation amounts are used to account for "washout" of some radionuclides during a rain event coinciding with the release. Therefore, precipitation shall be included in SLC 16.7-3 to ensure that this meteorological variable is demonstrated operable per the Testing Requirements.

Evaluation: The SLC/UFSAR changes that add the precipitation channel as a required meteorological constitutes a new technical requirement. The Meteorological System is not nuclear safety related, and does not perform any control, or accident mitigation functions. The system is not an accident initiator. The information provided from the precipitation channel is used as input into the RADDOSE-V program. The changes do not involve any safety or licensing issues. Regulatory Guide 1.23 Rev. 0 requires instrumentation for wind speed, wind direction, and air temperature at two different elevations on the same tower (delta T related to plume stability). RADDOSE V uses these inputs in its diffusion model for calculation of dispersion of radioactive materials. Precipitation is also used as an enhancement to the calculation. Precipitation is not a Reg. Guide 1.23 Rev. 0 requirement. An input of zero is acceptable, as this parameter was never used before. However, input is required for RADDOSE V to run.

The SLC change includes required instrumentation (Table 16.7-3A), surveillance requirements (Table 16.7-3B), and a clarification to the BASES.

Conservative (95%) X/Q (atmospheric dilution factors) values are used in Chapter 15 Accident Dose Analyses, which should bound most post accident emergency plan radiological assessments. The dose assessment efforts associated with the Emergency Plan are intended to reflect actual conditions. The Chapter 15 accident analysis dose calculations are not dependent on the Emergency Plan and there are no inputs to these calculations derived from the Emergency Plan efforts.

Reg. Guide 1.23 Revision 0 section C.5 specifies a minimum annual data recovery of 90%. However, there is no guidance given as to the reporting requirements to be taken if the 90% data recovery is not met. This SLC/UFSAR change will give administrative requirement to generate a Special Report to the NRC if the annual data recovery goal is not met.

There are no unreviewed safety questions associated with this Selected Licensee Commitment change. No Technical Specification changes are required. A change is

required to UFSAR Chapter 16.

229 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment 16.9-12 "Boration Systems Borated Water Sources-Operating" and SLC 16.9-11 Boration Systems Borated Water Sources-Shutdown

Description: Selected Licensee Commitment SLC 16.9-12 requires an Action and associated Completion Time for Refueling Water Storage Tank inoperablity (due to temperature or boron concentration not within limits) to make them consistent with Technical Specification 3.5.4. Technical Specification 3.5.4 allows 8 hours to restore temperature or boron concentration to within limits. SLC 16.9-12 was clarified to specifically address Refueling Water Storage Tank inoperability due to temperature or boron concentration to refer to LCO 3.5.4. ACTION A (8 hours) or proceed to shut down in compliance with LCO 3.5.4 ACTION C. Additionally, SLC 16.9-12 has been clarified to specifically address Refueling Water Storage Tank inoperability due to minimum water volume not within acceptable limits to refer to LCO 3.5.4.ACTION B (1 hour). Additional editorial discussion has been added to the SLC 16.9-12 Bases to identify the logic for the volume, temperature, and boron limits. Similar editorial references have been added to SLC 16.9-11 for the Shutdown Limits, and similar discussion has been added to the SLC 16.9-11 Bases.

Evaluation: There is no Unreviewed Safety Question associated with this change. No Technical Specification changes are required. No changes to the UFSAR are required to add to or modify the descriptions of station operating and shutdown procedures that are generally described in Chapter 13. No UFSAR changes are required in discussions regarding Chemical and Volume Control System boration functions and sources in sections 4.3.2.1, 5.4.7.2.6, 9.3.4.2.2.4, 9.3.4.2.3.2, 9.3.4.3. 1, and in the Chapter 15 (section 15.4.6) inadvertent boron dilution accident analyses. The fission product barriers of the pellet, clad, reactor coolant system pressure boundary and containment are not affected by this change to the SLC manual.

170 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 3-105 and Table 3-106

Description: There has been a manufacturer part number change for Solon pressure switch 7PS11DW to either 7PSW11D2 or 7PSW 11D1.

Modification CE-10365 added 7PSW11D2 to UFSAR Table 3-105 and Table 3-106. This modification was done as a result of an equipment substitution analysis. That analysis allowed 7PSW11D2 to be substituted for the 7PS11DW. A revision was made to the equipment substitution analysis in March 2000, which added the 7PSW11D1 model as an acceptable replacement for the 7PS11DW with ranges that were 0-3 INWC or less. Therefore, UFSAR changes will be made to Tables 3-105 and 3-106 to document the qualification of this new part number, 7PSW11D1.

Evaluation: The qualification method and test reference for Solon differential switches, models 7PS11DW and 7PSW11D2, are mentioned in UFSAR Table 3-105 and Table 3-106, Electrical Equipment Seismic Qualification for Catawba Unit 1 and 2, respectively. The new Solon model number, 7PSW11D1, will be added to these two tables. "Solon 7PS11DW" is also documented in Table 3-3 of Supplement 3 to the SER, NUREG0954 CNS SER, (not a revisable document). Replacing the Solon 7PS11DW (with a range of 0-3 INWC or less) pressure switch with a 7PSW11D1 that is identical in fit, form, and function is essentially a maintenance activity that meets the definition of an "equivalent component" found in department guidance documents.

There are no unreviewed safety questions associated with this UFSAR change. This change will have no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. UFSAR Tables 3-105 and 3-106 will be revised.

186 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 3-36 "Shell Wall Design Forces and Reinforcement"

Description: UFSAR Table 3-36 is being changed to correct the values of reinforcement required at the shell wall junction with foundation of the Reactor Building. These values were evaluated in engineering calculation CNC-1144.02-03-0001. The values given in Table 3-36 of UFSAR do not match with values evaluated in calculation CNC-1144.02-03-0001. The value of vertical steel in Table 3-36 is given as 6.61 square inches/ft (Fy = 40K), whereas in calculation this value is calculated to be 5.74 square inches/ft (Fy = 60K). Similarly the value of hoop steel given in Table 3-36 is 1.15 square inches/ft compared to 1.25 square inches/ft given in calculation CNC-1144.02-03-0001. Both values are within the values of reinforcement provided for shell wall, that is 6.24 square inches/ft for vertical steel and 1.56 square inches/ft for hoop steel.

Evaluation: Correcting the values of required reinforcing steel at the junction of the shell wall with foundation of the Reactor Building in Table 3-36 of the UFSAR does not constitute an unreviewed safety question. The error occurred while transporting values from a calculation to the table. Since the corrected values are from the original calculation and well within the reinforcement provided for the shell wall, the correction of reinforcing steel values in Table 3-36 of the UFSAR does not affect the design and safety of the shell wall of the Reactor building. No Technical Specification changes are required. UFSAR Table 3-36 will be revised.

187 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 3-49

Description: UFSAR Table 3-49 is being changed to correct the spacing of reinforcement given for the inside face at elevation 565+2" and the outside face at elevation 582+2" of the Crane Wall. The spacing of reinforcement was designed in calculation CNC-1144.03-11-0001 and is shown on design drawings CN-1050-60, CN-1050-61, CN-1050-62, CN-1050-66 and CN-1050-67. The correct spacings are # 11 at 1 degree-5 minutes instead of #11 at 1 degree-10 minutes for the inside face at elevation 565+2", and # 11 at 1 degree-10 minutes instead of #11 at 45 minutes for the outside face at elevation 582+ 2". This change has no implication on design and installation of reinforcement. This change revises the incorrect spacing given in Table 3-49 of UFSAR without affecting the design and safety of the Crane Wall.

Evaluation: Correcting the spacing of reinforcing steel given in Table 3-49 of UFSAR for inside face at elevation 565+2" and outside face at elevation 582+2" of Crane Wall does not constitute an unreviewed safety question. The error occurred while transporting data from a calculation to the table. The proposed change does not affect the design and analysis of the Crane Wall. No Technical Specification changes are required. UFSAR Table 3-36 will be revised.

202 **Type:** UFSAR Change **Unit:** 2

Title: UFSAR Change to Table 5-6, Primary Components Material Specifications

Description: UFSAR Table 5-6, "Class I Primary Components Material Specifications", contains several incorrect material specifications. Under the category of Steam Generator Components (Units 1 and 2), for Unit 2 Nozzle Safe Ends, the UFSAR states the material specification as "Stainless Steel Weld Metal Analysis Type 308L". This is incorrect and should be changed to "N/A", because there are no nozzle safe ends on Unit 2, only weld metal buildup.

Under the category of Steam Generator Components (Units 1 and 2). For Unit 2 Cladding and Buttering, the UFSAR states the material specification as "Stainless Steel Weld Metal Analysis A-7 and Ni-Cr-Fe Weld Metal F-Number 43". This is incorrect and should be changed to SFA 5.9 ER309L-SS" to agree with the material specified in the Westinghouse Model D-5 Steam Generator Stress Report (CNM-2201.01-0217). This is only a change to the UFSAR to reflect what material was actually used by Westinghouse.

Evaluation: These UFSAR revisions have no effect on the operation, design bases, or function of any structure, system or component. There are no Unreviewed Safety Questions associated with these changes. No Technical Specification changes are required. UFSAR Table 5-6 will be revised.

216 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 6-121

Description: UFSAR Table 6-121, "Refrigeration System Parameters", and UFSAR Sections 6.7.6.2, "System Design" and 6.7.6.3, "Design Evaluation" are being revised. The primary function of the ice condenser is the absorption of thermal energy released abruptly in the event of a loss-of-coolant accident, for the purpose of limiting the initial peak pressure in the containment. A secondary function of the ice condenser is the further absorption of energy after the initial incident, causing the containment pressure to be reduced to and held at a lower level for a period of time. The refrigeration subsystem serves as a central heat sink for ambient heat and heat of fusion picked up, respectively, in the ice condenser and in the ice machines. A circulating system of ethylene glycol solution carries the heat from the various heat transfer surfaces to the chiller packages. The Ice Condenser Refrigeration System is not nuclear safety related.

UFSAR Table 6-121 and the text of UFSAR Sections 6.7.6.2 and 6.7.6.3 will be revised. Currently, the table only lists the design parameters for the four Westinghouse Glycol Chilling Machines. Some additional Westinghouse design data will be added along with the design data for the two Carrier Glycol Chilling Machines. The description of the Carrier Glycol Chilling machines will be added to the above mentioned sections. Corrections will be made to design data for the Pressure Relief Valves and the Air Handling Units. The original design data for these components was provided by Westinghouse as proposed equipment design data, not as "as-built" data. The table has never been updated with the as-built data.

Evaluation: There is no unreviewed safety question associated with these UFSAR changes. These UFSAR changes involve revising the design parameters for the ice condenser refrigeration system. The UFSAR is changed to reflect the design data currently in design documents. The change of wording will not cause an accident evaluated in the UFSAR and it will not increase the probability of accidents evaluated in the UFSAR. This change will not affect the operation, design basis or function of any system, structure or component. No Technical Specifications changes are required. UFSAR Table 6-121and the text of UFSAR Sections 6.7.6.2 and 6.7.6.3 will be revised.

217 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 6-127 and Figure 6-175

Description: UFSAR Table 6-127 and UFSAR Figure 6-175 are being revised to add and correct information about the Ice Condenser Temperature Monitoring System. During normal operation, the Ice Condenser Temperature Monitoring System monitors the temperature of the ice bed, ice condenser floor, wall panels, and wear slab. Monitoring the temperature in the ice bed is required by a Technical Specification surveillance. Monitoring the other Ice Condenser Temperature Monitoring System temperatures is not required by Technical Specifications. The Ice Condenser Temperature Monitoring System does not function to mitigate design basis accidents and events. Items such as the elevation and radial location of individual instruments and the number of instruments are changed. None of the setpoints, design limits, or operating parameters of the Ice Condenser Temperature Monitoring System are changed. Currently, Table 6-127 contains numerous discrepancies regarding the number and placement of instruments in the Ice Condenser Temperature Monitoring System, and the information regarding the placement of instruments in Figure 6-175 is duplicated from Table 6-127. The changes and additions to UFSAR Table 6-127 and UFSAR Figure 6-175 will bring them into conformity with vendor drawings and design drawings, and remove information from Figure 6-175 that is duplicated in Table 6-127. A note will be added to Figure 6-175 to link it with Table 6-127, and a note will be added to Table 6-127 to link it with Figure 6-175.

Evaluation: There are no Unreviwed Safety Questions associated with this UFSAR change.The Ice Condenser Temperature Monitoring System does not function to mitigate any design basis accidents or events. During normal operation, the Ice Condenser Temperature Monitoring System functions to monitor ice condenser temperature, and ice bed temperature is an initial condition considered in safety analyses. Of five types or applications of instruments in the Ice Condenser Temperature Monitoring System, only the ice bed Resistance Temperature Detectors (RTD's) are used in Technical Specification surveillances. None of the other types or applications of instruments in the Ice Condenser Temperature Monitoring System have any relation to safety functions, design bases, or regulatory commitments. No Technical Specification changes are required. UFSAR Table 6-127 and UFSAR Figure 6-175 will be revised.

239 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 6-5 "Potential Water Traps Inside Containment"

Description: UFSAR Table 6-5 lists potential water traps inside containment and provides the approximate volumes of the water traps. During the process of reviewing of UFSAR Chapter 6, no basis for this table could be found. A calculation was originated to document the values in Table 6-5. The calculated values for potential trapped water are different from the values currently given in the table.

Evaluation: The changes to UFSAR Table 6-5 more accurately give the potential water traps in containment based on the results of calculation CNC-1223.21-00-0018. The volumes are not significantly different from what was in the table previously for three out of the four items. There is no water trap at the bottom of the refueling canal because there is a drain in the flat area at the lowest point. In any case the results of the table were not used for flood analysis at Catawba, but are in the UFSAR Table for information. The probability of an accident occurring at Catawba is not affected by the fact that slightly different values are given in UFSAR Table 6-5. There is no connection between accident initiation and the values in the Table. There is no unreviewed safety question associated with this UFSAR change. No Technical Specification changes are required. UFSAR Table 6-5 will be revised.

43 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 6-70

Description: The Design Pressure of 300 psig for the Containment Spray Pump appears to be a carryover from the McGuire UFSAR or the Westinghouse recommended rating. The McGuire Nuclear Station Containment Spray Pump is an Ingersoll-Rand and is rated at 300 PSIG. The Catawba Containment Spray Pump is manufactured by Bingham Willamette Co., and is only rated for 250 psig. The pump with the design pressure rating of 250 psig, was the pump that was tested to obtain satisfactory operating characteristics as described in CNM 1201.05-0039. The Containment Spray Pump discharge piping, per the Containment Spray system design parameter verification calculation CNC-1223.13-00-0001, is rated for 275 psig. Therefore the Containment Spray Pump Design Pressure should be changed from 300psig to 250psig.

Evaluation: There are no unreviewed safety questions associated with this UFSAR change. This correction does not involve any changes to the operation, design basis or function of any structure, system or component. No safety or licensing issues are involved and no revisions to regulatory commitments are involved by the corrections or changes. No Technical Specification changes are required. A change will be made to UFSAR Table 6-70.

240 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 6-77, "Containment Isolation Valve Data"

Description: The "Technical Specification 3.6.3 Applicable Condition" column in UFSAR Table 6-77 will be revised for the following containment penetrations on both Units: M317, M320, M330, M339, M343, M344, M350, and M351. The revision is necessary because the existing column entry incorrectly indicates that T.S. 3.6.3 Conditions A and B apply to these penetrations.

Evaluation: UFSAR Table 6-77 lists the design of these penetrations as conforming to 10 CFR50, Appendix A, Criterion 55 (Reactor Coolant Pressure Boundary Penetrating Containment). As such, each penetration must meet certain requirements for component configuration. These penetrations have a motor-operated valve(s) on the outside and a check valve on the inside. This arrangement is consistent with the methods described in ANSI N271(1976)/ANS 56.2 for meeting the requirements of General Design Criteria (GDC) 55. GDC-55 allows exceptions to the valve arrangements given if it can be shown that the containment isolation provisions of the new arrangement are acceptable on some "other defined basis". For the penetrations in question, the GDC requires the use of an automatic valve outside containment. Each penetration has outboard valves with electric motor operators, but none of these valves (with the exception of NV-314B in penetration M330) receive a signal (St, Sp) to close when containment isolation is needed. Under the definitions found in ANSI N271, these valves (with the exception of NV-314B) cannot be considered as automatic isolation valves. They more correctly fit the definition of remote manual valves. Therefore it must be shown, for the purpose of meeting GDC-55, that the use of remote manual valves is acceptable on some "other defined basis".

The "other defined basis" for these penetrations is as follows:

Containment isolation is achieved for penetrations M330, M339, M343, M344, M350, and M351 during design basis accidents (where containment source term is an issue) by the fact that one Centrifugal Charging Pump is always assumed running. None of the outboard containment isolation valves for these penetrations receive a signal to close. A single Centrifugal Charging Pump provides a pressure greater than peak accident pressure at each penetration. This pressure effectively seals the penetration and removes it from consideration as a leak path for the post accident containment atmosphere. Technical Specification 3.6.3 Actions A and B would not apply to these penetrations because the system design does not rely on the outboard containment isolation valves (CIV'(s)) closing in order to have containment isolation. The Technical Specification condition for one ECCS train inoperable is more appropriately entered if the outboard CIV('s) is determined to be inoperable.

For penetration M330 (Chemical and Volume Control System Charging Line), sealing pressure is always provided by at least one Centrifugal Charging Pump. Both valves NV314B and NV312A receive a signal to close on a safety injection. Valve NV314B is the credited CIV. Should NV314B fail to close, a Centrifugal Charging Pump and redundant valve NV312A assure containment isolation. Since failure of one of these valves to close is a flow diversion issue rather than a containment isolation issue. The Technical Specification for one ECCS train inoperable is more appropriately entered if one of the outboard CIV's is determined to be incapable of closing

An analysis was performed for penetrations M320 and M317 (Safety Injection System to Hot Legs) in conjunction with removing the Containment Penetration Valve Injection Water System from outboard CIV's NI-121A and NI-152B. The analysis shows that containment isolation is provided by a positive sealing pressure on the outboard valve at all times post accident when the penetration is not in service. The Technical Specification for one ECCS train inoperable is more appropriately entered if one of the outboard CIV's is determined to be inoperable.

There is no unreviewed safety question associated with this UFSAR change. No Technical Specification changes are required. UFSAR Table 6-77 will be revised.

201 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 6-77, Unit 1 and 2 Containment Isolation Valves Data, UFSAR Figure 6-112, Figure 6-113, Figure 6-114, Figure 6-115 Containment Piping Penetration Arrangements

Description: UFSAR Table 6-77 "Unit 1 and 2 Containment Isolation Valve Data" and associated UFSAR Figures 6-112, 6-113, 6-114, and 6-115 "Containment Piping Penetration Arrangements" are being revised. The revisions fall into the following groups:

1. Correcting the selected valve arrangement drawing. This information is being corrected to agree with the figures in the UFSAR.

2. Correcting referenced UFSAR figure numbers. This information is being corrected to agree with the figures in the UFSAR.

3. Correcting and/or adding valve sizes and nominal line sizes. This information is being corrected to agree with the figures in the UFSAR.

4. Correcting flow direction. This information is being corrected to agree with the flow convention as stated in the table notes (15).

5. Correcting whether on not seismic equipment is connected to inside and/or outside of containment. This information is being corrected to agree with the figures in the UFSAR.

6. Correcting valve types. This information is being corrected to agree with the figures in the UFSAR.

7. Correcting and or adding component numbers. This information is being corrected to agree with the figures in the UFSAR.

8. Correcting and/or adding notes. This information is being corrected to agree with the figures in the UFSAR.

9. Correcting actuator types. This information is being corrected to agree with the figures in the UFSAR.

10. Correction actuation signal. This information is being corrected to agree with the figures in the UFSAR.

11. Correcting and/or adding normal position of valves. This information is being corrected to agree with the figures in the UFSAR.

12. Correcting and/or adding shutdown position of valves. This information is being corrected to agree with Operation Procedures. These procedures have been reviewed per the 10CFR 50.59 USQ Evaluation Process.

13. Correcting and/or adding post accident position of valves. This information is being corrected to agree with the figures in the UFSAR.

14. Correcting and/or adding fall safe position. This information is being corrected to agree with the figures in the UFSAR.

15. Correcting valve location as to whether it is inside or outside containment. This information is being corrected to agree with the figures in the UFSAR.

16. Correcting whether or not Type C tested. This information is being corrected to agree with Appendix J Test Procedures and Program.

17. Correcting type of test required (B or C). This information is being corrected to agree with Appendix J Test Procedures and Program.

18. Correcting whether penetration is drained for type A test or not. An annotation to see Note 52 is being added to every page header. This information is being corrected to agree with Appendix J Test Procedures and Program.

19. Correcting Applicable TS 3.6.3 condition. This change is being corrected to agree with TS 3.6.3. Applicable Conditions.

20. Deleting superfluous information, correcting typographical errors. This information is given in other Sections, Figures and/or Tables in the UFSAR and is not needed in this Table, or is an editorial typographical error.

21. Deleting type and size of vent and/or drain valves. This information is given in other Sections, Figures and/or Tables in the UFSAR and is not needed in this Table.

Evaluation: There are no unreviewed safety questions associated with these UFSAR changes. These or changes do not involve any changes to the operation, design basis, or function of any structure, system or component. No Technical Specification changes are required. UFSAR Table 6-77 and UFSAR Figures 6-112, 6-113, 6-114, and 6-115 will be revised.

233 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 7-4

Description: UFSAR Table 7-4 does not completely agree with the UFSAR Chapter 15 discussions regarding reactor trip correlations. UFSAR Chapter 15 accidents have been identified that credit certain trip functions, but are not listed under that trip function in Table 7-4. A review has shown that not all of the reactor trip correlations discussed in Chapter 15 have been listed in the appropriate locations within UFSAR Table 7-4. Calculation CNC-1552.08-00-0317 justifies markups to UFSAR Table 7-4 such that there is complete agreement between this table and UFSAR Chapter 15.

Evaluation: There are no Unreviewed Safety Questions associated with this change. Revising this table will have no effect on the probability or consequences of accidents evaluated in the UFSAR. No Technical Specification changes are required. UFSAR Table 7-4 will be revised.

150 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 8-3

Description: UFSAR Table 8-3 is being revised to correct the conductor size of the 230KV Roddey transmission line. The conductor type will be changed from "Single 954 KCM 54/7 ACSR" to "Bundled 954 KCM 54/7 ACSR". This change will bring the UFSAR into agreement with the current transmission system design drawings. This change does not involve any physical changes to the plant.

Evaluation: There are no unreviewed safety questions associated with this UFSAR change. This UFSAR change has no effect on the operation, design basis, or function of any plant equipment or structure. These changes are the result of revisions to design drawings depicting the 230KV electrical transmission system. No Technical Specification changes are required. A change is required for UFSAR Table 8-3.

71 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 9-3

Description: UFSAR Table 9-3 is being revised to reflect actual system flow conditions which were measured after modifications CN-11248 and CN-20639 were performed. After these modifications were implemented, the Containment Chilled Water System flow rates to the Upper Containment Ventilation Units (UCVUs) were balanced to prevent overcooling of the upper containment if a loss of the Instrument Air System supply air occurred and the control valves failed open. Procedure PT/1(2)/A/4400/12 was used to balance flow rates to the UCVUs. The new cooling water flow rates were not incorporated onto flow diagrams and other system documents at the time. The proposed changes to be made by the current modification will incorporate nominal flow rates set under the old modifications. The Containment Ventilation System design basis document will be revised to show the nominal field flow rates. The Nuclear Service Water System design basis document will be revised to remove statements which indicate that the Containment Coolers maintain containment temperatures within limits during design basis events since this is not true in any of the safety analyses or calculations associated with these events. UFSAR Table 9-3 will be revised to approximate the Nuclear Service Water System flow rates to the UCVUs based upon the Containment Chilled Water System flow adjustments made after modifications CN-11248 and CN-20639 were implemented. UFSAR Table 9-3 is also being revised to show the Lower Containment Ventilation Unit nominal flows (800 gpm) measured in PT/1(2)/A/4400/11. The changes performed per this modification are editorial in nature.

Evaluation: The Containment Ventilation System does not provide any safety related function and is not required to mitigate the consequences of any postulated accidents. Its function of containment temperature control is for normal plant operation and normal shutdown only. The LOOP Containment Ventilation System design basis function will not be adversely affected by this modification. Technical Specification 3.6.5 describes the limits for containment temperatures. This Technical Specification is not affected by this modification. The Nuclear Service Water System is used during Loss of Offsite Power to provide cooling water to the containment cooling systems since the Containment Chilled Water System will not be available. The UCVU flow rates revised in UFSAR Table 9-3 and the Nuclear Service Water System Design Basis Document are estimated values based upon the current plant configuration. No credit is taken for these flow rates in any safety analyses to mitigate any design basis event. The changes made to the UCVU flow rates will not affect cooling of the reactor coolant pumps upon a loss of the Containment Chilled Water System System. The proposed changes to the UFSAR, Design Basis Documents, Test Acceptance Criteria Sheets, and Flow Diagrams for the Containment Ventilation System, Containment Chilled Water System, and Nuclear Service Water System serve to clarify the description of present operating conditions already in place. The configuration of the plant was modified under Modifications CN-11248 and CN-20639 and the changes to the plant were evaluated under the associated 10CFR50.59 evaluations for those modifications. The documents which will be modified under this modification, CE-70113, should have been changed per Modifications CN-11248 and CN-20639 but were apparently overlooked.

The Containment Ventilation System and the Containment Chilled Water System function together to maintain acceptable temperature limits within the confines of the Reactor

Building upper and lower compartments to ensure proper operation of equipment and controls during normal plant operation and normal shutdown and for personnel access during inspection, testing, and maintenance. The Containment Ventilation System does not provide any safety related function and is not required to mitigate the consequences of any postulated accidents. Its function of containment temperature control is for normal plant operation and normal shutdown only. Nuclear Service Water System water supply to the Containment Ventilation system occurs during loss of offsite power since Containment Chilled Water System cooling water is not available.

The Containment Ventilation System and the Containment Chilled Water System maintain temperatures within the established containment environmental qualification limits and initial average temperatures assumed in the DBA LOCA and SLB analyses. This modification is only correcting plant documents to reflect the current Containment Chilled Water System configuration. This modification will not adversely affect the design basis function of the Nuclear Service Water System/ Containment Ventilation System and the Containment Chilled Water System or the temperature limits associated with Technical Specification 3.6.5.

There are no unreviewed safety questions associated with this UFSAR change. No Technical Specification changes are required. UFSAR Table 9-3 will be revised.

76 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to UFSAR Figure 1-20

Description: Modification CN-50466 transferred all of the meteorological instruments from the 40-meter microwave tower and 10-meter low level meteorological tower to a new 60-meter tower. As part of that modification, a change to UFSAR Figure 1-20, "Plot Plan" was determined to be necessary. However Figure 1-20 was not revised. Another modification (CE-70098) was created to facilitate the change to Figure 1-20.

Evaluation: This UFSAR change has no effect on the operation, design bases, or function of any structure, system or component. There is no effect on the probability or consequences of accidents evaluated in the UFSAR. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. UFSAR Figure 1-20 will be revised.

33 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Figure 7-14

Description: An editorial change is being made to UFSAR Figure 7-15. There is a figure continuation note at the bottom of the diagram which refers to "Fig. 7.6.6-2" which is incorrect. The figure/page numbers were previously revised and these continuation references were overlooked during that change. The correct continuation reference is to Figure 7-16.

Evaluation: This change has no effect on the probability or consequences of accidents evaluated in the UFSAR. The change is editorial in that it corrects an item which was overlooked when UFSAR Figures were renumbered to support a different software program. No Technical Specifications changes are required. UFSAR Figure 7-14 will be revised.

23 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Figure 7-26

Description: UFSAR Figure 7-26 provides a basic diagram of the flux-mapping system beginning with the drive units, through the 5-path and 10- path transfers, and through the thimble tubes into the reactor vessel. The flux-mapping system has isolation valves at the seal table which are not shown on Figure 7-26. A review of plant documents has shown that there are thimble tube isolation valves at the seal table. Plant drawing CNM 2399.06-0001-005, provides a diagram of the detector drive train showing the thimble tube isolation valves at the seal table. A note should be added to UFSAR Figure 7-26 clarifying that the isolation valves are non-safety-related and are not considered in the design basis accident analysis.

Evaluation: There are no unreviewed safety questions associated with this UFSAR change. Changing UFSAR Figure 7-26 to include the existing isolation valves and aadding a clarifying note does not change any assumptions or inputs to the safety analyses. The change will have no effect on the probability or consequences of accidents evaluated in the UFSAR. No changes to the Technical Specifications are required. UFSAR Figure 7-26 will be revised.

221 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Figure 7-6

Description: UFSAR Figure 7-6 is being changed to correct a note on the figure. The note currently states "From Turbine Driven AFW Pump Logic (Figure 7.4.1-2)". This is a figure numbering scheme that was used before converting the UFSAR to an electronic format in the mid 1980's. This figure number was never changed to the new numbering scheme . The reference should state: "From Turbine Driven AFW Pump Logic (Figure 7-7)."

Evaluation: There is no unreviewed safety question associated with this UFSAR change. The change will have no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. UFSAR Figure 7-6 will be revised.

25 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Section 10.4.9.2

Description: UFSAR Section 10.4.9.2 will be revised to reduce the minimal Auxiliary Feedwater System Temperature from 40 degrees F. to 32 degrees F. This temperature change is supported by a reanalysis of the UFSAR Chapter 15 steam line break accident analysis that was performed by the Safety Analysis Group. The reanalysis is documented in calculation CNC-1552.08-00-0076 revision 6.

Evaluation: The Auxiliary Feedwater System is designed to "provide a nuclear safety related source of emergency feedwater to the steam generators to maintain secondary side level, at times when the normal feedwater system is not available". The system "is relied upon to remove primary coolant stored energy and residual core energy, and to prevent over pressurization of the Reactor Coolant System and the resultant reactor coolant expansion that could result in fuel damage". The system is designed to supply cooling water to the "S/G nozzles in the pressure range from the Residual Heat Removal System cut-in conditions (equivalent to approximately 110 psig S/G secondary side pressure) to the relieving pressure of the lowest safety relief valve (1210 psig)". UFSAR Section 10.4.9.2 currently lists a supply temperature range of 40 degrees F. to 138 degrees F. The minimum temperature is listed as 60 degrees F. in the Safety Analysis Inputs Manual and 32 degrees F. in Table 6-66 of the UFSAR. The Safety Analysis Group has reanalyzed the affected Chapter 15 Steam Line Break Accident Analysis for a minimal Auxiliary Temperature of 32 degrees F. The reanalysis is documented in CNC-1552.08-00-0076, rev 6. No problems were identified with the analysis, and the Auxiliary Feedwater System will continue to be able to perform as required in response to a main steam line break.

The reanalysis was combined with the reanalysis to increase the minimum reactor coolant system flow rate limit. The increase in minimum reactor coolant system flowrate was in support of Catawba Nuclear Station, Units 1 and 2 Technical Specification Amendment Number 176 and 184, respectively, These amendments were approved by the NRC on March 1, 2000. The reanalysis to lower the Auxiliary Feedwater System temperature did not change any values in the Unit 1 and 2 Technical Specifications (although, the reanalysis was bundled with the increase in RCS temperature which required NRC approval).

This revision to section 10.4.9.2 of the UFSAR to lower the minimum Auxiliary Feedwater Temperature supply does not create any unreviewed safety questions. The reanalysis performed by the Safety Analysis Groups shows that the Auxiliary Feedwater System will continue to meet the design requirements to support the UFSAR Chapter 15 analysis for a main steam line break.

16 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Section 6.2.5.7, Supplemental Hydrogen Control System/Hydrogen Mitigation System/Hydrogen Ignition System

Description: UFSAR Section 6.2.5.7 includes a description of the "Igniter Power Supply". This discussion states that the igniters are powered from the Class 1E "emergency lighting" power system. The section should state "A/C" rather than "emergency lighting".

Evaluation: There is no unreviewed safety question associated with this UFSAR change. The change revises incorrect information. This change has no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. UFSAR Section 6.2.5.7 will be revised.

24 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Section 7.2.1.1.2 and UFSAR Figure 7-3

Description: UFSAR Figure 7-3 provides a generic diagram of the Overtemperature Delta-T Trip setpoint reduction function. The title of Figure 7-3 does not include the Overpower Delta-T Trip. Figure 7-3 does not contain any reference to the fact that the Unit 1 Overtemperature Delta-T Trip "negative" flux imbalance penalty breakpoint and slope are not applicable. A review of the Core Operating Limits Reports (COLR) and the CNEI-0400-84, Rev. 1, identified that the Unit 1 "negative" flux imbalance penalty for Overtemperature Delta-T does not apply. A note will be added to Figure 7-3 referencing the COLR for constants and breakpoints which will account for the differences between Unit 1 and Unit 2 "negative" flux imbalance penalty for Overtemperature Delta-T. UFSAR Figure 7-3 is a generic drawing that can be used to represent both the Overtemperature and the Overpower delta-T Trip setpoint reduction function. UFSAR Section 7.2.1.1.2, item b, "Overpower Delta-T Trip" equation, (flux imbalance description) in the description list of equation components should include a reference to Figure 7-3 similar to the Overtemperature Delta-T equation.

In Section 7.2.1.1.2 - Reactor Trips, item b, second item in the description list of equation components will be changed by adding a reference to Figure 7-3 as follows: (Refer to Figure 7-3)"

Figure 7-3 " Setpoint Reduction Function for Overtemperature delta-T Trips", Insert the words "Overpower and" in front of the word "Overtemperature" in the title of Figure 7-3. Add a note stating: "NOTE: See Core Operating Limits Reports (COLR) for constants and breakpoints."

Evaluation: Correcting this UFSAR Figure does not constitute an unreviewed safety question. The correction has no effect on the probability or consequences of accidents evaluated in the UFSAR. No Technical Specification changes are required. UFSAR Section 7.2.1.1.2 and UFSAR Figure 7-3 will be revised.

22 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Section 7.7.2

Description: UFSAR Section 7.7.2 provides a discussion of the analysis of the plant control systems design for high reliability in any anticipated operational occurrences. A discussion is provided of the review of IE Bulletin 79-27, "Loss of Non-Class 1E Instrumentation and Control Power System Bus During Power Operation, Actions 1 through 3," and how Catawba complies. A description of the automatic functions of the non-1E auxiliary control inverters includes the following: "Relative to non-Class 1E non-safety-related power supply inverters: 1) The non-Class 1E auxiliary control inverters employ a non-adjustable 5-10 second time delay which is only used to delay an automatic transfer from the alternate source back to the inverter and prevents the transfer until the inverter has synchronized with the alternate source." A review of plant documents has shown that this is not entirely correct. Plant procedure IP/0/A/3540/001 "Corrective Maintenance Procedure SCI Inverters," performs a functional test of the inverters static switch. In a procedure step, synchronization between the alternate source and the inverter is initiated and static switch swap (transfer) between the alternate source and the inverter within 60 seconds is verified. Vendor manual CNM 1358.02-0008-001, "Instruction & Operating Manual; Inverter Model SV 12300/TSNB/TSMB," Section 4A, Parts List No. 14913 for the static switch and drawing CNM 1358.02-0006-001, "Parts List No. 14913, Single Phase Static Switch, 30 KVA with Auto-Retransfer" both identify relay RL205 with a time delay of 1-60 seconds. Drawing CNM 1358.020003-001, "Schematic-30 KVA, 1-Pole Static Switch with Auto-Retransfer" provides a schematic of the static switch using relay RL205. Deleting the text "non -adjustable 5-10 second" does not change any assumptions or inputs to the safety analyses. Increasing the delay of the automatic transfer from the alternate source back to the inverter until synchronization is complete is necessary to protect the equipment. Increasing the time delay by use of a 1-60 second time delay is a good engineering practice used to ensure synchronization between power sources and would also aid in preventing cycling of the load if the inverter did not respond properly.

In UFSAR Section 7.7.2 paragraph two, the sentence; "Relative to non-Class IE non-safety-related power supply inverters: 1) The non-Class 1E auxiliary control inverters employ a non-adjustable 5-10 second time delay which is only used to delay an automatic transfer from the alternate source back to the inverter and prevents the transfer until the inverter has synchronized with the alternate source" will be revised to state; "Relative to non-Class 1E non-safety related power supply inverters: 1) The non-Class 1E auxiliary control inverters employ a time delay which is only used to delay an automatic transfer from the alternate source back to the inverter and prevents the transfer until the inverter has synchronized with the alternate source."

Evaluation: UFSAR text will be changed to reflect the actual inverter static switch circuit configuration, based on a review of procedure IP/0/A/3540/001, "Corrective Maintenance Procedure SCI Inverters," Manufacturer's Drawing CNM 1358.020006-001, Parts List No. 14913, "Single Phase Static Switch, 30 KVA with Auto-Retransfer," CNM 1358.02-0003-00 1, "Schematic-30 KVA, 1-Pole Static Switch with Auto-Retransfer", and vendor manual CNM 1358.02-0008-001, "Instruction & Operating Manual; Inverter Model SV12300/TSNB/TSMB. There are no unreviewed safety questions associated with this UFSAR change. No changes to the Technical Specifications are required. UFSAR section

7.7.2 will be revised.

31 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Section 9.2.5.3.1

Description: UFSAR Section 9.2.5.3.1 is being revised to change the description of the results of calculations performed by Duke Power Company. These calculations confirm the ability of the Standby Nuclear Service Water Pond to provide cooling water to the plant for up to 30 days in the event of simultaneous Loss of Coolant Accident (LOCA) on one unit, Loss of Offsite Power (LOOP) on both units, Loss of Lake Wylie, and shutdown of the non-LOCA unit. The calculations have been refined to more accurately reflect the performance of the Nuclear Service Water System and the Standby Nuclear Service Water Pond. Specific changes to the calculation include a more detailed listing of the heat inputs to the pond, corrections to the analysis of the Nuclear Service Water System to more accurately reflect the flow capacity of the system, changes to the most severe meteorological conditions calculation, and inclusion of results of the Duke Power Gothic Program for calculation of the decay heat from the reactor core on the LOCA unit. Previously the decay heat input to the calculation was taken from curves supplied by Westinghouse. Duke has taken responsibility for these calculations and performs them in-house. In all cases, the results of these calculations provided more conservative results. The heat input to the Standby Nuclear Service Water Pond was increased as a result of these calculations and the cooling water flow rate was decreased. The results showed that the Standby Nuclear Service Water Pond is adequately sized to support the plant under these most severe conditions.

Evaluation: This UFSAR change has no effect on the operation, design bases, or function of any structure, system or component. These changes do not increase the probability or the consequences of any accidents previously evaluated or the possibility of an accident of a different type than previously evaluated in the UFSAR. These changes do not increase the probability or consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR or the possibility of a malfunction of a different type than previously evaluated. No Technical Specifications changes are required. UFSAR Section 9.2.5.3.1 will be revised.

32 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Section 9.2.5.3.3

Description: The Catawba Updated Final Safety Analysis Report (UFSAR) Section 9.2.5.3.3, "Hydraulic Design", (Ultimate Heat Sink Section), includes a description and the results of Duke Calculation CNC-1150.01-00-0001, Standby Nuclear Service Water Pond - Thermal Analysis During One Unit LOCA and One Unit Shutdown. The methodology previously used in this calculation had been disputed by the NRC. The NRC and Duke Power disagreed on the appropriate methodology for calculating the ability of the Standby Nuclear Service Water Pond to supply water to Catawba for a maximum of 30 days in the event of a simultaneous LOCA and Loss of Lake Wylie and concurrent shutdown of the unaffected unit. As a compromise solution, this calculation was revised using a new computer program. This new calculation resulted in slightly different models being used with slightly different results being obtained. In all cases, the new methodology and calculation results are more conservative than previously used. This change to the UFSAR corrects the description in the UFSAR to agree with the new methodology and results obtained by calculation CNC-1150.01-00-0001, revision 12 dated August 24, 1999.

Evaluation: This UFSAR change has no effect on the operation, design bases, or function of any plant structure, system or component. This change does not increase the probability or the consequences of any accidents previously evaluated or the possibility of an accident of a different type than previously evaluated in the UFSAR. These changes do not increase the probability or consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR or the possibility of a malfunction of a different type than previously evaluated. No Technical Specifications changes are required. UFSAR Section 9.2.5.3.3 will be revised.

220 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Section 9.3.1.1

Description: This change to UFSAR Section 9.3.1.1 will add references to Figures 9-226, 9-227, 9-228, 9-229, 9-230, 9-231, 9-232, 9-234, 9-235, 9-236, 9-237, 9-238, 9-239, 9-240, 9-241, 9-242, 9-243, 9-244, 9-245, 9-246, 9-247, 9-248, 9-250, 9-251, 9-252, 9-253, and 9-254 to the description of Section 9.3.1.1. These figures are flow diagrams for the Instrument Air System and the Station Air System. The flow diagrams are already in the UFSAR. This change adds a reference to them in Section 9.3.1.1.

Evaluation: There is no change to either the Instrument Air System or the Station Air System associated with this UFSAR change. This UFSAR change adds a reference in the text of Section 9.3.1.1 to refer to these flow diagrams. This UFSAR change has no effect on the probability or consequences of accidents analyzed in the UFSAR. There are no unreviewed safety questions associated with this UFSAR change. No Technical Specification changes are required. UFSAR Section 9.3.1.1 will be revised.

172 **Type:** UFSAR Changes **Unit:** 0

Title: Selected Licensee Commitment 16.9-8 Boration System Flow Paths - Operating and Selected Licensee Commitment 16.9-10 Boration System Charging Pumps - Operating

Description: Pursuant to 10 CFR 50.90, Duke Energy requested temporary changes to Technical Specification 3.5.2, Emergency Core Cooling System (ECCS) - Operating, Technical Specification 3.6.6, Containment Spray System (CSS), Technical Specification 3.6.17, Containment Valve Injection Water System (CVIWS), Technical Specification 3.7.5, Auxiliary Feedwater (AFW) System, Technical Specification 3.7.7, Component Cooling Water (CCW) System, Technical Specification 3.7.8, Nuclear Service Water System (NSWS), Technical Specification 3.7. 10, Control Room Area Ventilation System (CRAVS), Technical Specification 3.7.12, Auxiliary Building Filtered Ventilation Exhaust System (ABFVES), and Technical Specification 3.9.1 AC Sources - Operating for Catawba Nuclear Station Unit 2. The proposed Technical Specification changes will allow the "A" and "B" Nuclear Service Water System (NSWS) headers to be sequentially taken out of service for twelve days each for cleaning and pipe replacement. This cleaning and pipe replacement is scheduled to occur when Unit 1 is in refueling outage 1EOC12 and Unit 2 is at power operation. As a result of these changes, Selected License Commitment 16.9-8 "Boration Systems Flow Paths - Operating" and Selected License Commitment 16.9-10 "Boration Systems Charging Pumps - Operating" are being changed in a manner similar to these NRC approved Technical Specification changes. This 10CFR50.59 evaluation reviewed these Selected License Commitment changes using the same logic and justification applied to the Technical Specification changes and no unreviewed safety questions were discovered. Therefore, Selected License Commitment 16.9-8 and Selected License Commitment 16.9-10 will be revised accordingly.

Evaluation: There are no Unreviewed Safety Questions associated with these Selected Licensee Commitment revisions. No Technical Specification changes are required. No UFSAR changes are required.