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April 1, 2000

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
1999 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 1999. 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

Attachment

IE47

xc:

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Units 1 and 2

1999 10CFR50.59 Report

April 1, 2000

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 1999. The entries are organized by the type of activity being evaluated in the following order:

185 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-05079, Abandon the Floor Drain Leak Detection System

Description: Minor Modification CE-05079 will abandon the Floor Drain Leak Detection System. The individual leak detectors will be removed from their drains and discarded whenever practical. If area dose rates are high, associated items will be abandoned in place. All control power wll be disconnected such that the system will be de-energized. Stainless steel drain covers will be glued to the drain openings.

Evaluation: There are no unreviewed safety questions associated with this minor modification.The system is not nuclear safety related. The probability or consequences of accidents evaluated in the UFSAR will not be affected by this modification. No new accident scenarios will be created. No Technical Specification changes are required. UFSAR Section 1.8.1.34.15 will be revised.

6 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-09033, Replace obsolete Source/Intermediate Range Assembly WL-23821 with NY-10034

Description: Minor Modification CE-09033 will replace obsolete Source/Intermediate Range Assembly Part Number WL-23821 with Part Number NY-10034. This is a form, fit, and function replacement for an item that is no longer available.

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

69 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-09278

Description: Minor Modification CE-09278 replaces an air flow transmitter for one of the Lower Containment Ventilation Units in the Unit 1 Containment Building. A new database record will be created to list the transmitter and airflow monitoring device as separate instruments where previously the transmitter and air flow device was combined into a single instrument record.

Evaluation: The Containment Ventilation System is not nuclear safety related and no credit has been taken for operation of the system in analyzing the consequences of accidents. The transmitter is not required for seismic integrity, and adding the new transmitter will not affect any seismically designated structure, system, or component. The instrumentation loop for the affected transmitter is shown on Figure 9-211 of the UFSAR. This figure will be changed to reflect the new transmitter instrument number. The transmitter will not affect the operating characteristics of the Ventilation System. Failure of the new transmitter will not affect the ability of structures, systems, or components to function as required by the UFSAR.
This modification does not involve any unreviewed safety questions. No changes are required to the Technical Specifications. UFSAR Figure 9-211, a piping flow diagram, will need to be revised to show the new instrument number for the transmitter.

41 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-09287

Description: A concern was identified about the amount of aluminum allowed inside containment as stated in the UFSAR. The HEPA filters in the Containment Auxiliary Charcoal Filter Units (CACFU's) were found to be made of aluminum which caused the original containment aluminum limit to be exceeded. A re-analysis of the allowable amount of aluminum that could be in containment was performed. This re-analysis demonstrated that with all the aluminum accounted for inside containment (including the HEPA filters), the hydrogen mitigation systems would still be operable. However, it was decided to remove the CACFU's HEPA filters during Modes 1 through 4 and leave the original aluminum limit stated in the UFSAR unchanged. Since the CACFU's HEPA filters were removed during Modes 1 through 4, CE-9287 was written to revise the appropriate documentation. The Containment Ventilation System and the Containment Chilled Water System function together to maintain acceptable temperature limits within the Reactor Building upper and lower compartments to ensure proper operation of equipment and controls during normal plant operation and normal shutdown as well as for personnel access during inspection, testing, and maintenance. This System does not provide any safety related function and is not required to mitigate the consequences of any postulated accident. The CACFU's are only used for personnel contamination protection, so having the HEPA filters removed in modes 1 through 4 will not have any effect on the function of the system. In addition, having the HEPA filters removed except when needed will not affect any manual/automatic features, introduce any unreviewed system interactions, alter any seismic or environmental qualifications, affect the quality group of the CACFU's, or affect core reactivity. The HEPA filters are not used in accident mitigation and are not safety related. There would be no increase in radiological dose to the public during an accident due to the removal of the HEPA filters. The HEPA filters are only installed in Modes 5, 6, or No Mode when airborne activity levels reach a point that require cleanup of the containment atmosphere for personnel doing maintenance, testing, or inspections.

Evaluation: This activity will not change, degrade, or prevent SSC's from operating nor prevent actions from occurring that are necessary to mitigate any accident described in the SAR. None of the assumptions used to evaluate the radiological consequences of an accident are affected by this activity. Access to accident mitigation equipment is unaffected by the filter removal. Also, no fission product barriers are jeopardized as a result of this change in operation of the CACFU's. This modification involves no Unreviewed Safety Questions. No changes to the Technical Specifications are required. UFSAR Figure 9-131 will be revised to document the removal of the CACFU HEPA filters in Mode 1 through Mode 4.

60 Type: Minor Modification **Unit:** 1

Title: Minor Modification CE-09509, Perform various actuator replacements and MOV testing

Description: Minor Modification CE-09509 involves various motor operated valve actuator replacements and MOV testing. 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: 1BB057B, 1BB060A, 1BB147B, 1BB148B, 1BB149B, 1BB150B, 1CA046B, 1CA054B, 1KC320A, 1KC332B, 1KC424B, 1KC429B, 1KC430A, 1NC056B, 1ND001B, 1ND002A, 1ND032A, 1ND036B, 1NF233B, 1NI009A, 1NI010B, 1NI076A, 1NI088B, 1NI095A, 1ND096B, 1NI118A, 1NI120B, 1NI122B, 1NI154B, 1NI162A, 1NI332A, 1NI333B, 1NM003A, 1NM006A, 1NM007B, 1NM025A, 1NM026B, 1NS012B, 1NS029A, 1NV089A, 1NV091B, 1NV189B, 1NV312A, 1NW105B, 1RF389B, 1RF447B, 1RN404B, 1SM074B, 1SM076B, 1VQ003B, 1VS054B, 1WL450A, 1WL451B, 1WL805A and 1WL807B. Meeting the requirements of Generic Letter 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 setup data. New thrust or torque set-up windows are being established to increase each MOVs margin for operation.
In addition, new actuators are being installed on 1ND032A, 1NI118A, 1NS012B, 1NS029A, 1RF389B, 1RF447B, and 1RN404B and new motors are being installed on 1NI076A and 1NI088B. These modifications are also being performed to increase each MOV's margin for operation.
The spring pack for valve 1KC429B is also being replaced with a Limitorque Model 0301-109 and the valve item number will be changed to 09J-624. Additional changes being made by this modification are adding an anti-hammer contact on valve 1NV189B and modifying the hangar on valve 1RF389B.

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 the modification. No new failure modes have been introduced as a result of this modification. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. Changes to UFSAR Figures (Piping Flow Diagrams) are required.

7 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-09540, Replace butterfly valve 1WP018

Description: Minor Modification CE-09540 will replace the manually operated butterfly valve which is currently installed at tag location 1WP018. The currently installed butterfly valve (valve item 02B-420) is corroded and leaking and is no longer useful as an isolation valve.

The replacement butterfly valve item number is CMV-664. Valve item 02B-420 is a "lugged" butterfly valve (it bolts directly to a flange). Valve item CMV-664 is a "wafer" butterfly valve. Thus, to "wafer" valve item CMV-664 into place, a second flange will be added to the downstream side of the replacement butterfly valve. Engineering has reviewed and approved this change. Valve 1WP018 is the "Recirculated Cooling Water Pump Sump Drain to Unit 1 Recirculated Cooling Water Valve Pit Sump Valve". All of the applicable butterfly valve parameters associated with this change-out have been evaluated per the valve replacement evaluation form which was prepared for the modification.

Flow diagram CN-1604-2.00 will be revised. The flow diagram currently shows an extension stem installed at tag location 1WP018. There is no extension stem installed at tag location 1WP018 and there is no provision for an extension stem associated with valve item 02B-420. There is no extension stem associated with valve item CMV-664. Consequently, the extension stem notation on the subject flow diagram will be removed.

Evaluation: The "fit, form and function" of this non-nuclear safety related tag location 1WP018 will not be affected by this modification. Consequently, the proposed activity will not increase the probability of occurrence of an accident previously evaluated in the SAR. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. No UFSAR changes are required.

120 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-09726, Replace valve 1NC28 and 1NC30 with new valves, DMV-1079

Description: Minor Modification CE-09726 will replace valve 1NC28 and 1NC30 with new valves, DMV-1079. Valves 1NC28 and 1NC30 are the Reactor Coolant System Loop A and Loop B Pressurizer Spray Control Bypass Throttle Valves. In January 1998 it was determined that the C Heaters were not able to maintain pressurizer pressure. Investigation determined that this was caused by seat leakage past these valves. In addition to the leakage problem, the valves have had operational problems. The valves are Kerotest needle valves, which means that the disc and stem are separated by metal diaphragm. Since there is no direct connection between the disc and stem, the valve can be opened without the disc actually moving. This makes it difficult to determine that the valves are properly closed or opened. For these two reasons the valves will be replaced with new globe valves, item number DMV-1079.

Evaluation: The new valves are considered functionally equivalent replacements. Any differences between the new valves and the previously installed valves have been evaluated and found to be acceptable. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. UFSAR Figure 5-3, a piping flow drawing, will be revised.

32 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-10021, Replace two inch Auxiliary Feedwater System Tempering Flow Piping with Chrome-Moly Material

Description: Minor Modification CE-10021 replaces the Unit 1 Auxiliary Feedwater System Tempering Flow two inch socket weld piping with Chrome-Moly (P-11) Material. This change is being made due to the effects of flow accelerated corrosion. The modification only addresses a piping material change. No other system parameters are affected. The replacement material is an approved ASME Section III material and is acceptable for use in the Auxiliary Feedwater System.

Evaluation: There are no unreviewed safety questions associated with this modification. All pipe rupture concerns have been addressed and all replacement materials meet the applicable code requirements. No Technical Specification changes are required. A UFSAR change will be made to the associated piping flow diagram.

8 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-10052, Replace Valve 1RN278

Description: This Minor Modification will replace valve 1RN278 (Item Number 06J-501) with new Item Number 02D-701. Presently this valve is a one inch Y-Type Globe valve with internal damage. It will be replaced with a one inch ball valve, which will continue to serve the same function. All affected drawings will be revised to reflect this new information. Valve 1RN278 is the "1B Auxiliary Building Supply Ventilation Unit Drain Valve". (RN = Nuclear Service Water System).

Evaluation: There were no Unreviewed Safety Questions associated with this modification. Replacement of valve 1RN278 with an equivalent component is a maintenance activity that is not addressed in any manner in the SAR. This is not a significant plant change that would require inclusion in the SAR. This Minor Modification involves a one-for-one component replacement/improvement. The modification will result in enhanced system operability and availability. 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. No Technical Specification changes are required.

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

Title: Minor Modification CE-10095, Delete barriers between Essential Switchgear Rooms and Electrical Penetration Rooms from the scope of NRC committed fire boundaries

Description: Minor Modification CE-10095 will delete barriers between Essential Switchgear Rooms and Electrical Penetration Rooms from the scope of NRC committed fire boundaries. Each of these rooms contains equipment associated with the same train of electrical equipment. A fire in either area would result in reliance on the redundant train of electrical equipment that is located on a separate elevation of the Auxiliary Building to achieve and maintain safe shutdown.

Evaluation: The removal of these walls from the committed fire barriers does not result in an unreviewed safety question. Changing this boundary will not affect the probability that fire will occur. A fire in either area would result in the same safe shutdown scenario. This scenario is loss of the associated train of safe shutdown equipment with shutdown from the control room using the opposite train components. A fire in either room would not affect the operability of the opposite train equipment which is located on another elevation of the Auxiliary Building. No Technical Specification changes are required. No UFSAR changes are required. Selected Licensee Commitment 16.9.5 will be revised to reflect this change.

10 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-10107, Jumper out temperature switch 0YCTS9209B in 2CRA-C-1

Description: This modification will install a jumper to bypass the Control Room Area Chiller (2CRA-C-1) low temperature load recycle switch (0YCTS9209B) contacts in the chiller control circuitry. The chilled water load recycle switch has malfunctioned and no replacement is available.

The chilled water load recycle switch (0YCTS9209B) is a one of several equipment protection instruments installed on the Train B Chiller (2CRA-C-1). The chilled water load recycle switch functions to prevent the chiller evaporator water tubes from freezing by automatically cycling the chiller compressor on/off during low cooling load conditions.

The chiller has redundant equipment protection features. Therefore, elimination of the low chilled water load recycle switch for 2-CRA-C-1 will not adversely impact the operation of the chiller during normal or design basis accident conditions. The chiller has another evaporator tube freeze protection feature in the low refrigerant temperature cutout switch (0YCTS9208B). The low refrigerant temperature cutout switch is designed to stop the chiller compressor when the refrigerant temperature reaches 33 degrees F. After the low refrigerant temperature cutout switch opens to stop operation of the chiller, the switch must be manually reset to restart the chiller. Therefore, evaporator tube freeze protection will still exist without the low chilled water load recycle switch.

The ability of the chiller to maintain a chilled water supply temperature necessary for the system air handling units to maintain ambient air temperatures below the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system will not be impacted. Installation of the jumper within the chiller control circuitry will not adversely impact any emergency diesel generator load sequencer start signal.

Evaluation: There are no unreviewed safety questions associated with this minor modification. The chiller has other protection features which will perform the same equipment protection feature. No Technical Specification changes are required. No UFSAR changes are required.

235 **Type:** Minor Modification **Unit:**

Title: Minor Modification CE-10119, Delete Filtered Water Flow Chart Recorder and modify water flow instrumentation to provide a signal to the Turbidity Chart Recorder

Description: Minor Modification CE-10119 will perform changes to the Filtered Water System Flow instrumentation to address a failed pneumatic chart recorder in the Filtered Water System. The scope of this modification includes deleting the recorder, installing a pneumatic-to-electric transducer, routing the output of the transducer to a spare channel of the turbidity chart recorder which is on the same panel. Several minor documentation changes will be made as well.

Evaluation: The Filtered Water System functions to remove suspended solids from lake water to produce filtered water acceptable for use at the station. The Filtered Water System water acts as a source of supply water to the following:

Makeup Dernineralized Water System
Condenser Circulating Water Pump Oil Cooler and Bearings
Coagulant Mixing and Storage Tank
Seal Water for various vacuum equipment and organic biocide
Chilled Water for the Hypochlorite Generator
Priming Water for the Turbine Lube Oil Purifiers
Bearing Flush Water for the Service Building Sump Pumps
Interior Fire Protection System
Makeup Water to the Auxiliary Building Cooling Water System

There is no unreviewed safety question associated with this modification. The Filtered Water System does not perform a nuclear safety function and is not required during any design basis accident. Filtered Water System flow indication is provided on a chart recorder mounted on a local panel. The affected instrumentation is located in an area that does not require seismic mounting of equipment and components. Although this instrumentation is not directly mentioned in the UFSAR, it is shown on UFSAR System flow diagrams. No Technical Specification changes are required. UFSAR Figures 9-50 and 9-51 will be revised.

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

Title: Minor Modification CE-10130, Revise Diesel Generator Jacket Water Low Pressure setpoint to reflect as-built conditions

Description: Minor Modification CE-10130 revises Diesel Generator Jacket Water Low Pressure setpoint to reflect as-built conditions. This modification provides an editorial change to the Diesel Generator vendor manual to show the correct setpoint for low jacket water system pressure. It was discovered that the manual which shows this information had not been revised when this setpoint change was originally made in 1984.

Evaluation: This change has no effect on the ability of the Diesel Generator to perform its safety function. There is no unreviewed safety question associated with this modification. No Technical Specification changes are required. A change is required for UFSAR Table 9-40.

12 Type: Minor Modification **Unit:** 0

Title: Minor Modification CE-10134, Revise the Containment Purge Ventilation System and Contaiment Hydrogen Sample and Purge System Design Basis Documents to reflect implememtation of Improved Technical Specifications and Revisions to the UFSAR

Description: Minor Modification CE-10134 will revise the Containment Purge Ventilation System and Contaiment Hydrogen Sample and Purge System Design Basis Documents to reflect implementation of Improved Technical Specifications and revisions to the UFSAR. Implementation of the Improved Technical Specifications (ITS) resulted in containment isolation valve data such as isolation and Engineered Safety Features (ESF) response times being transferred from the old Technical Specifications to UFSAR Tables 6-77 and 7-15.

The Containment Hydrogen Sample and Purge System containment isolation valve (CIV) isolation times in UFSAR Table 6-77, "Unit 1 and 2 Containment Isolation Valve Data" and the Containment Purge Ventilation System and Containment Hydrogen Sample and Purge System ESF response times in Table 7-15, "ESF Response Times" were revised. The containment isolation and ESF response times for these containment isolation valves were changed to "Not Applicable" in UFSAR Tables 6-77 and 7-15 because the valves are sealed or locked closed during Modes 1, 2, 3, and 4. In addition, a note was added to UFSAR Table 9-29, 'Purge System Isolation Valve Design and Test Criteria", to clarify that testing of the Containment Hydrogen Sample and Purge System CIV closure times is not performed because the isolation valves are sealed or locked closed during Modes 1, 2, 3, and 4.

Evaluation: Each Unit 1 and 2 Containment Purge Ventilation System contains nine containment penetrations and each penetration contains two redundant containment isolation valves. During normal plant operations, these valves are administratively locked closed by de-energizing their solenoid valves. The valves are only opened during cold shutdown and refueling activities.

The Containment Purge Ventilation System containment isolation valves are assumed to be open during a postulated fuel handling accident within the containment and no credit is taken for their closure during the accident. The radiological consequences of a postulated fuel handling accident are acceptable because the release path includes a non ESF filter train tested in accordance with Regulatory Guide 1.52 and the lack of containment pressurization potential during refueling mode 6.

Each Unit 1 and 2 Containment Hydrogen Sample and Purge System contains two containment penetrations and each penetration contains two redundant containment isolation valves. Three of these valves are motor operated gate valves with soft seats and one is a passive check valve. During normal plant operations, the motor operated gate valves are administratively locked closed by de-energizing their actuators. The passive check valve located inside the containment maintains a closed position since the blower is not placed in operation.

The Containment Hydrogen Sample and Purge System containment isolation valves are only opened during cold shutdown or no mode activities. The Containment Hydrogen Sample and Purge System containment isolation valves are maintained in a closed position during refueling activities.

These containment penetrations with their associated piping and valves are nuclear safety related. The design basis function of the Containment Purge Ventilation System and Containment Hydrogen Sample and Purge System containment isolation valves is to maintain containment integrity and limit radiological doses during a design basis accident such as a LBLOCA. Since the Containment Purge Ventilation and Containment Hydrogen Sample and Purge System containment isolation valves are sealed closed during Modes 1, 2, 3, and 4, they are in their design basis ESF closed positions during normal plant operations and prior to initiation of any design basis accident. Therefore, the Containment Purge Ventilation and Containment Hydrogen Sample and Purge System containment isolation and ESF response times in UFSAR Tables 6-77 and 7-15 are insignificant. A note was placed in each table to indicate that the Containment Purge Ventilation and Containment Hydrogen Sample and Purge System containment isolation valve stroke and ESF response times are "Not Applicable" since the valves are locked closed during Modes 1, 2, 3, and 4. Locked closed is the same as sealed closed. A note was also added to UFSAR Table 9-29 to show that testing of the Containment Purge Ventilation System CIV closure times is not performed because the isolation valves are sealed or locked closed during Modes 1, 2, 3, and 4. Information was also added to the Containment Purge Ventilation and Containment Hydrogen Sample and Purge System design basis documents to reflect these changes to the UFSAR Tables.

10CFR50 Appendix J Type C leak rate testing is periodically performed to ensure overall containment leakage is within specified l imits during Modes 1, 2, 3, and 4. Each Containment Purge Ventilation and Containment Hydrogen Sample and Purge System containment isolation valve is also verified to be in its closed position once every 31 days. These surveillances ensure that containment integrity will be maintained prior to and during any design basis accident.

These UFSAR and DBD changes will not result in any changes to the normal operation of the plant. The Containment Purge Ventilation and Containment Hydrogen Sample and Purge System containment isolation valves are sealed closed during Modes 1, 2, 3, and 4 and will be in their design basis ESF closed positions during normal plant operations and prior to initiation of any design basis accident during these modes. These valves cannot initiate any design basis accidents. Therefore, these changes for the Containment Purge Ventilation and Containment Hydrogen Sample and Purge System containment isolation valves will not increase the probability of occurrence of an accident previously evaluated in the SAR. There are no unreviewed safety questions associated with this modifcation and UFSAR change. No Technical Specification changes are required. Changes are required for UFSAR Table 6-77, Table 7-15 and Table 9-29.

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

Title: Minor Modification CE-10147

Description: Minor Modification CE-10147 corrects a flow diagram to show valve 1RNE32 in the normally open position and revises the Nuclear Service Water System Design Basis Specification (DBD) to state the correct Nuclear Service Water System and Recirculated Cooling Water System valve lineups used during a manual alignment of the Nuclear Service Water System to the Instrument Air System compressors. The Nuclear Service Water System, including Lake Wylie and the Standby Nuclear Service Water Pond (SNSWP), is the ultimate heat sink for various QA Condition 1 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 the Nuclear Service Water System Pumphouse, and cooling flow and flush water for non-QA heat loads and functions during normal operation. 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. The majority of pneumatic operated Nuclear Service Water System components are on the nonessential headers and fail open upon loss of offsite power or loss of instrument air. During normal plant operation, if instrument air is lost to any or all non-safety pneumatically operated valves, the effect will be identical to a Loss of Offsite Power Event whereby cooling water flow to the nonessential components will be maximized. The Operators are relied upon to start the remaining nuclear service water pumps to prevent pump runout. Should the loss of instrument air occur concurrent with a Design Basis Event, system operation will not be affected since the nonessential headers are isolated. A loss of offsite power design basis event may occur by itself or concurrent with any other design basis event. Since the Nuclear Service Water System provides cooling water to the diesel generators, the Nuclear Service Water System is required to operate during a loss of offsite power event. The Nuclear Service Water System is aligned to Instrument Air Compressors E and F during a loss of offsite power event to maintain the non-safety related compressors operable to prevent the consequences of a loss of instrument air event. Valve 1RNE32 is in the return line for the Nuclear Service Water System from the Instrument Air Compressors and is required to be open for this return flow path. To eliminate having to open 1RNE32 as part of the Nuclear Service Water System to Instrument Air System lineup, this valve is left in the normally open position. Valve 1RNE32 was added during the implementation of Minor Modification CE-3638 to isolate the Containment Chilled Water System from the operating portion of the Nuclear Service Water System during some piping replacement. This valve was never intended to be left closed and station procedures were changed to reflect that this valve was in the open position following completion of modification CE-3638.

Evaluation: None of the accidents or equipment important to safety as described in the SAR are affected by this activity since the Nuclear Service Water and Instrument Air systems will not operate any differently due to 1RNE32 being shown normally open on the flow diagram. This modification involves no Unreviewed Safety Questions. No changes to the Technical Specifications are required. UFSAR Figure 9-28 (Nuclear Service Water System flow diagram) will need to be revised to show valve 1RNE32 in a normally open position.

158 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-10167, Replace Meteorological Monitoring Systrem Ambient/Delta Temperature Module

Description: This modification will replace Meteorological Monitoring System Ambient/ Delta Temperature Processor Module. The installed module is a Teledyne Geotech Model 40.35 Temperature Processor, which is obsolete and has reached the end of its service life. This module will be replaced with a Met-One Module 21.320-14 Temperature Processor and a Model 21.321-12 Delta Temperature Processor. UFSAR Section 2.3.3 will be changed to reflect the change in manufacturer, Selected Licensee Commitment 16.7-3 will be changed to reflect the new calibration requirements for the new modules. An additional change will be made to clarify that both the Ambient and Delta Temperature Channels are required for system operability. Calibration Procedure IP/0/B/3343/013 "Meteorological Monitoring System Calibration and Maintenance Procedure", will be revised to reflect the new calibration methods. The new modules will have the same form and perform the same function as the original module; however, the old model was a single unit that occupied two slots in the equipment rack. The new modules are two separate units which will occupy the same two slots in the equipment rack. Minor equipment rack wiring changes will be required and a Duke Energy manufactured test panel will be installed to facilitate testing of the temperature channels.

Evaluation: The meteorological system is not nuclear safety related and does not perform any control, accident prevention or accident mitigation function. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. Changes are required for UFSAR Section 2.3.3 and the Selected Licensee Commitments (UFSAR) Section 16.7-3.

71 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-10201 Replace valve 1WLD48 with a new valve, CMV-691

Description: Minor Modification CE-10201 will replace valve 1WLD48 with a new valve, CMV-691. The currently installed valve is a Nupro globe/needle valve that has been clogging. The new valve will be a ball valve which should eliminate clogging concerns associated with the globe valve design. This valve is a manual isolation valve to the radwaste sample sink in the Liquid Radwaste System.

Evaluation: The new valve is considered equivalent in form, fit, and function to the previously installed valve. The differences between the two valves have been evaluated and found acceptable. This valve does not serve any function which would affect the probability or consequences of those accidents evaluated in the UFSAR. There are no unreviewed safety questions associated with this change. No Technical Specification changes are required. A change will be made to the system flow drawing in the UFSAR.

81 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-10283, Revise ice basket drawings to clear up ambiguities in instructions and show proper configuration with repair alternatives

Description: Minor Modification CE-10283 allows field work in the Ice Condenser to take place in the future and documents a potential existing situation for configuration control purposes.

Current Ice Condenser ice basket drawings do not reflect the exact configuration of some hardware options installed or planned for upcoming maintenance work. The components that are considered for incorporation in this modification are covered in three areas:
1. Ice basket assembly sheet metal screw
   (a) Installation
   (b) Relocation of basket assembly screws as a repair option
2. Ice basket cable cruciform suspension system top plate J-bolts variation.
3. Ice basket drawing (CNM 1201.17-0030 and CNM 1201.17-0436) editorial enhancements

An additional ice condenser drawing related to the top deck blankets will be updated to reference the proper nut material.
Top deck blanket drawing (CNM 1201.17-0512) editorial enhancements will also be made.

These problems are described in detail below:
(1) Ice basket assembly sheet metal screw:
   (a) Installation - Instructions presently permit the use of self tapping screws as repairs for screws broken out of baskets, or in replacement baskets at their installation. Screws affected by this modification are not self drilling as original equipment, but were supplied by the original equipment manufacturer. A note on drawing CNM 1201.17-0030, sheets 1, 9, 11 forbids pre-drilling pilot holes. The drawings (and sheet 13) need to be revised to permit drilling pilot holes for self tapping screws.
   (b) Relocation of basket assembly screws as a repair option - Minor Modification CE-9791/00, allowed for relocation of screws due to broken screws or stripped hole threads. The previous modification applies only to top basket ring sheet metal screws. The Detail of Alternate Self Tapping Screw Connection, needs to apply to other couplings and bottom assemblies in addition to top rings (same detail, notes 1, 2 and 3 do not apply).
(2) Ice basket cable cruciform suspension system top plate J-bolt variation - Two variations of the J-bolt are in use presently in the ice condenser. Only one variation is represented on drawing CNM 1201.17-0030 sheet 4. Corrective Action Program Report 0-C99-1367, describes a J-bolt with an approximate 30 degree bend midway in the shank, and is not represented on drawing CNM 1201.17-0030 sheet 4. J-bolts have been purchased in several different lots, from different manufacturers and to slightly differing specifications. The bent J-bolts were purchased to the same material strength and test requirements as

other lots.

(3) Ice basket drawing editorial enhancements - Some drawing details cause time consuming questions and confusion to contract workers and inspectors. The drawing problems include details of components that are no longer used or stocked, add references to obscure related drawings and make notes and bill of material item numbers consistent. These changes involve drawings CNM 1201.17-0030 sheets 1, 3, 4, 5 and 7.

(4) Top deck blanket drawing editorial enhancements - Drawing bill of materials (BOM) cause time consuming questions and confusion when ordering replacement parts fasteners. The drawing BOM problems include calling nuts by the same ASTM designation as bolts, which is not technically accurate or consistent. These changes involve drawing CNM 1201.17-0512 sheet 1.

Modifications are outlined as follows:

1. Ice basket sheet metal screw
   (a) Installation - Revise drawings to permit predrilling pilot holes through the basket shell and coupling/support ring/top ring bottom assembly, in order to start the self threading screws. The hole size is based on ANSI B18.6.-1981, for a self tapping screw into 14 gauge sheet steel with # 10-32 threads. The hole size is recommended to provide adequate thread engagement between the self tapping screw and the basket wall. The drill size is based on the basket wall section thickness. The screws are supplied by Westinghouse, the OEM. Replacement screws are necessary because: repairs to basket top rings are needed periodically, new screws are used when basket sections are replaced, the pre-drill information was not previously furnished as original screws were self drilling and self tapping.

   (b) Relocation of basket assembly screws as a repair option - Revise drawings CNM 1201.0030 sheets 1, 9 and 11, to permit relocating screw holes in the same radial centerline, but circumferentially a minimum of 1 inch from the existing hole in either direction. Modification CE 9791/00, permits this relocation in this manner for top rings.

   Westinghouse was consulted to assure their concurrence.

2. Ice basket cable cruciform suspension system J-bolt variation - Revise the affected drawing, to permit use of J-bolts with a specified bend in the shank, and minor other dimensional differences. The purchase order under which the bent J-bolts were furnished required testing of a sample of J-bolts to assure adequate strength. Receipt documentation supplied with the shipment, states that testing was performed on a sample of the manufactured bolts, and the test results were satisfactory. This is the same testing required for all J-bolts supplied for this application.

3. Ice basket editorial drawing enhancements - Change drawings CNM 1201.17-0030 sheets 1, 5, 7, 9, 11, 13 and CNM 1201.17-0436 sheet 2 to simplify information and references.

4. Top deck blanket drawing editorial enhancements - changes involve drawing CNM 1201.17-0512 sheet 1 to correct fastener reference specification designations. The drawings reference bolting specifications for nuts, which is no longer acceptable.

Evaluation: The ice condenser functions as a passive device, wholly contained (for its safety function) in the containment, for automatic initiation in the event of a design basis accident, a high energy line break in containment. The ice condenser serves to:

(1) Limit the pressure increase in containment by condensing steam contact with ice.
(2) Furnish a large amount of water, which when melted drains directly into the containment sump.
(3) Remove fission product iodine from the atmosphere during ice melting and subsequent spray via the containment spray system.

The ice condenser is an annular compartment enclosing approximately 300 degree azimuth of the perimeter of the containment building, extending through the divider deck into both the lower and upper containment. The ice condenser is designed and constructed to permit steam pressure in excess of one pound per square foot to open the lower inlet doors and pass through the ice bed. Sufficient pressures and flows will also open intermediate deck doors and top deck blankets. The ice baskets and supporting structures maintain the requisite amount of ice in the optimum geometry for transfer of heat from steam to ice. Because of sublimation phenomenon, methods and techniques had to be developed to replace ice mass lost to the ice condenser environment. Ice replenishment and other hardware replacement programs necessitated modification of basket hardware to allow an efficient and timely ice mass replenishment process, within existing space and containment penetration limitations. The sheet metal screws serve to hold ice basket coupling rings, support rings, top rings and bottom assemblies in place. Couplings assemble one basket section to another. Bottom assemblies provide the connection between ice baskets and the lower support structure. Couplings, bottom assemblies and basket sections are designed to resist design basis accidents, Safe Shutdown Events and dead weight loads. The sheet metal screws have the same material properties as the original equipment screws and are furnished by the original equipment manufacturer. The ice basket cable cruciform suspension system was developed and qualified to give the plant options in ice mass replenishment and replaceable ice supports (cruciforms). J-bolts are part of the cable cruciform suspension system and serve to hold down the star (top) plate against any uplift of the ice column resulting from a design basis accident. Ice column uplift is the result of dynamic forces pushing up on the bottom and sides of ice columns, tending to eject the ice column from the basket. Total uplift load is approximately 3029 pounds force, with (6) J-bolts sharing the uplift load. If each J-bolt carried an equal load, each bolt would restrain approximately 505 lbs. A sample of the J-bolts are tested in tension to 2500 lbs. each, with acceptance criteria: no yielding or cracking. Thus each J-bolt has a proven safety factor of approximately 5.0.

The ice condenser is physically separated from any areas of the containment where an accident could occur or be caused to occur. The components and structures involved in this minor modification serve no function during normal operation of the plant and provide no normal operating support to plant functions or activities. Because of separation and lack of any relationship to normal plant operation, there is no increase in probability of an accident because of this minor modification.

The modification presents optional installation methods but there is no reduction in load carrying capability of the components affected. Since the sheet metal screws and J-bolts develop the same loads as UFSAR values, the modification in the ice condenser here cannot increase the probability of occurrence of a malfunction of such equipment. The probability of a malfunction of equipment in the ice condenser will not be increased. This modification does not change the passive nature or capability of the ice condenser.

The ice condenser will still function as designed to perform its mitigation function in a design basis accident. This is acceptable based on the modification related fasteners developing full load carrying capability for basket connections, with screw installation methods and J-bolts as stated above. Further, the J-bolts in their alternate configuration were proven by sample test to have the same load carrying capability as those represented on the CNM 1201.17-0030 drawing sheet 4.

There is no malfunction of equipment associated with the minor modification evaluated here since the loads do not change and the fasteners develop the same design load carrying capability as the original equipment. The ice baskets and ice condenser will perform as designed with the minor modification in place. Containment recirculation and other Emergency Core Cooling System (ECCS) functions will not be impacted by the modification.

There is no physical connection or credible possibility of an interaction between ice condenser baskets and any system or equipment involved in normal operation of the station including reactor pressure boundary or containment components. There is no creditable possibility for the creation of a different type of accident than those evaluated in the SAR.

There is no type of malfunction of equipment that is created by the minor modification here, as the loads do not change and the fasteners develop the same load carrying capability as the original equipment design. The ice baskets, ice condenser, ECCS, and Containment Spray System will still perform as designed taking into account this modification, and no different type of malfunction of equipment is introduced.

None of the parameters related to the operation of the ice condenser are impacted by this minor modification. No Technical Specifications directly related to the ice condenser, ECCS, and Containment Spray System are impacted by this modification. Tests and inspections described in the UFSAR Chapter 6 and design basis documents are not impacted by the minor modification described here.

The minor modification evaluated here revises vendor supplied drawings to provide an alternate connection installation method and screw location options for the ice condenser basket couplings, bottom assemblies and rings, and alternate J-bolt configuration for the cable cruciform suspension system star (top) plate. The alternate connection allows the use of self tapping, but not self drilling, sheet metal screws by predrilling a pilot hole. Included is relocation of screw holes if a screw is broken or threads stripped. The sheet metal screw installation has been agreed to by Westinghouse (the OEM and supplier of the screws). The use of alternate J-bolt configuration for the cable cruciform suspension system permits an equally tested fastener to be used with a slight bend in the shank. The tested tensile load provides a safety factor of approximately 5.0 when loaded in the same manner as installed in the ice basket. This modification revises drawings to permit both the above installation and configuration options. The modification further revises ice basket drawings to enhance and simplify information and references, which has proven to be a problem to contract workers and inspectors in ice condenser maintenance and ordering replacement fasteners. These changes allow the ice condenser to function as designed with no actual change to the structure or to performance in a design basis accident. There are no unreviewed safety questions associated with this modification. No changes are needed in the UFSAR. UFSAR Chapter 6 and Chapter 15 Safety Review and Design Bases Sections remain unchanged. The fission product barriers of the fuel pellet, cladding, Reactor Coolant System primary pressure boundary and containment are not affected at all by this modification.

119 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-10285

Description: This modification will revise the Containment Ventilation System Design Basis Document to reflect current operations procedures which allow versatility in operation of the Upper and Lower Containment Ventilaton Units. The current system procedure allows all units to be started with one unit in standby. When the procedures were revised it was determined that the UFSAR did not require a change since the system description was general. This modification does propose a UFSAR change which is considered an enhancement. This modification also deletes some information from the Design Basis Document which does not pertain to the system and clarifies the description of certain instrumentation and equipment. This change is editorial.

Evaluation: This change to the UFSAR and the Design Basis Document for the Containment Ventilation System clarifies the description of current operating practices already in place. The existing procedures allow any combination of Upper Containment Ventilation Units and Lower Containment Ventilation Units to be operated during normal operation.

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 any accident evaluate in the UFSAR. Its function is only for temperature control during normal plant operation and normal plant shutdown. There is no unreviewed safety question associated with this minor modification. No Technicall Specification changes are required. A change will be made to UFSAR Section 9.4.6.

95 Type: Minor Modification **Unit:** 0

Title: Minor Modification CE-10338

Description: Minor Modification CE-10338 involves changes to the design basis document for the Containment Air Return and Hydrogen Skimmer System, and changes to procedures PT/1/A/4200/009 (Revision 165N) and PT/2/A/4200/009 (Revision 140O). Operability concerns were identifed with the Hydrogen Skimmer System suction valves Engineered Safety Features (ESF) response times. The ESF Response time for the suction valves 1(2)VXlA and 1(2)VX2B as stated in procedure PT/1(2)/A/4200/09, "Engineered Safety Features Actuation Periodic Test", were greater than the allowable time listed in UFSAR Table 7-15 "ESF Response Times". It was determined that these valves are only required to start opening (i.e. they did not have to be fully open) within 8 - 10 minutes after a Containment High High Pressure (Sp) Signal. In order to clarify the design basis function of the Hydrogen Skimmer System suction valves, modification CE-10338 will revise the Containment Air Return and Hydrogen Skimmer System design basis document, CNS-1557.VX-00-0001. CNS-1557.VX-00-0001 will be revised to clarify that the valves only start opening after the 8 - 10 minute time delay and on-site power is supplied through the Emergency Diesel Load Sequencer System Load Group 1. Additional information will be included to explain the Hydrogen Skimmer System suction valve ESF response times.

UFSAR Table 7-15 will be revised and Technical Specification 3.6.8 Bases will be revised to make the required changes.

PT/1(2)/A/4200/09 will also be revised to reflect the correct Hydrogen Skimmer System suction valve full open ESF Response Times. These changes will be included in Revisions 165N and 140O for Units 1 and 2 respectively.

Evaluation: The initial design basis function of the Hydrogen Skimmer System suction valves is to prevent bypassing the ice condenser during the initial blowdown phase of a large break loss of cooling accident . The valves perform this function by remaining in their closed position during the early stages of a design basis accident. After a 9 ± 1 minute (480 to 600 seconds) delay, the suction valves start to open and the Hydrogen Skimmer Fans start to reduce hydrogen concentrations by drawing from dead end spaces within the lower containment. Hydrogen pocketing is prevented by continuously drawing air from the dead end spaces at a rate that limits the potential local hydrogen concentration to less than 4% by volume. The Hydrogen Skimmer Fans discharge the air near the Hydrogen Recombiners. The Hydrogen Recombiners are manually placed in operation within 24 hours of a design basis LOCA to ensure the containment atmosphere remains below 4% hydrogen concentration.

CNS-1557.VX-00-0001, Containment Air Return & Hydrogen Skimmer System Design Basis Specification, is being revised to reflect actual system operation as described in UFSAR Section 6.2.1.1.3. 1, the Bases for Technical Specification SR 3.6.8.3, and the Containment Air Return and Hydrogen Skimmer System electrical elementary drawings will be revised as well. The electrical elementary drawings show that a Hydrogen Skimmer Fan start is enabled as soon as the suction valve actuator limit switch moves off the closed position. Information was also added to the Containment Air Return and Hydrogen Skimmer System DBD to show that the suction valves are on Diesel Load Sequencer Load Group 1. These changes are being performed to reflect actual system operation and documentation. Therefore, these changes are considered editorial.

A detailed explanation of the Hydrogen Skimmer System suction valve ESF Response Times will also be added to the Design Basis Document. These details are explained below. In UFSAR Table 7-15, Note 9, will be added to clarify the ESF Response Times for the Containment Air Return and Hydrogen Skimmer System Operation as well as the Hydrogen Skimmer System suction valves. Item 5.d of UFSAR Table 7-15 currently lists 600 seconds as the maximum response time for the Containment Air Return and Hydrogen Skimmer System Operation. The 600 seconds refers to the operation of the Containment Air Return and Hydrogen Skimmer Fans used to satisfy Surveillance Requirement 3.3.2.10 for the Containment Air Return and Hydrogen Skimmer System. UFSAR Table 7-15 Note 9 will clarify that the Hydrogen Skimmer System suction valves should be fully open after 668 seconds for a LOCA and 679 seconds for a LOCA/Blackout. These suction valve ESF Response Times support the way the Hydrogen Skimmer System was designed to perform its design basis function of maintaining hydrogen concentrations less than 4% by volume. Each Hydrogen Skimmer Fan and associated suction valve has a separate time delay set at a nominal value of approximately 540 seconds (9 minutes). After the nominal 540 second time delay is reached and other permissives are satisfied, the Hydrogen Skimmer System suction valve starts to open and the Hydrogen Skimmer Fan starts (Technical Specification 3.6.8). Since the suction valve's full stroke time is 66 seconds maximum per design (PT/1(2)/A/4200/36), it is apparent that the valves were never designed to be fully open within 600 seconds. These changes will not adversely impact Technical Specifications 3.3.2 (SR 3.3.2.10) or 3.6.8.

This change to UFSAR Table 7-15 will not adversely impact operation of the Hydrogen Skimmer System as described in the UFSAR. UFSAR Section 6.2.1.1.3.1 states that the Hydrogen Skimmer Fan is designed to start once the suction valve starts to open. The Containment Air Return and Hydrogen Skimmer System electrical elementary drawings also support Hydrogen Skimmer Fan startup after the suction valve starts to open. Additional licensing basis documents were reviewed and it was concluded that the 600 seconds ESFAS Response Time for the Containment Air Return and Hydrogen Skimmer System Operation did not include requiring the suction valves to reach their full open position.

CNC-1552.08-00-0194, Revision 3, "Reanalysis of the Catawba Hydrogen Skimmer System Flow Requirements", was issued to evaluate hydrogen concentrations within the containment after a 720 second (12 minute) time delay of Hydrogen Skimmer System operation. The results indicate that the Hydrogen Skimmer System is capable of performing its design basis function of maintaining hydrogen concentrations less than 4% by volume with a 12 minute time delay and the Containment Air Return and Hydrogen Skimmer System initiation time is not considered a controlling parameter in the analysis.

UFSAR Section 6.2.5.3.1 states, "The results of this analysis demonstrate that as long as the hydrogen recombiner is placed in service by 24 hours following the design basis LOCA, the hydrogen concentration in Containment will not exceed 4 volume percent". This supports the fact that bulk hydrogen concentration is not a concern during the first 24 hours after a design basis accident.

Therefore, allowing the Hydrogen Skimmer System suction valve ESF Response Times to be 668 (2 + 600 + 66) seconds for the LOCA and 679 (2 + 11 + 600 + 66) seconds for the LOCA/Blackout in UFSAR Table 7-15 will not adversely impact the ability of the Hydrogen Skimmer System to perform it's design basis functions. The 2 seconds accounts for Solid State Protection System (SSPS) instrument delay time and is consistent with the assumptions in UFSAR Section 7.3. The 11 seconds is the maximum ESF Response Time for emergency diesel generator operation upon a safety injection signal (UFSAR Table 7-15). The 600 seconds is the maximum time delay for Hydrogen Skimmer Fan startup (SR 3.3.2.10 and 3.6.8.4). The 66 seconds is the maximum suction valve stroke time per PT/1(2)/A/4200/36. The Hydrogen Skimmer System suction valve ESF Response Times are documented in calculation CNC-1223.02-00-0004, "Response Time Testing Requirements".

The Background Section of the Technical Specification 3.6.8 Bases will be revised to correctly match Hydrogen Skimmer System operation as described in the Bases for Surveillance Requirement (SR) 3.6.8.3. This is considered an editorial change because it reflects current information in UFSAR Section 6.2.1.1.3.1, the Containment Air Return and Hydrogen Skimmer System electrical elementary drawings, and the Bases for SR 3.6.8.3.

The current PT/1(2)/A/4200/09 suction valve acceptance criteria is 664.8 seconds, for the LOCA/Blackout and LOCA test sections. Based on the UFSAR Table 7-15 discussion above, the procedure acceptance criteria should be 666.8 seconds for a LOCA and 677.8 seconds for a LOCA/Blackout to account for diesel generator startup and SSPS instrument error uncertainty. The 1.2 second ESF Response Time variation from UFSAR Table 7-15 in these procedures accounts for instrumentation and electrical response time and Operator Aided Computer scan time.

These Design Basis Document, UFSAR, Technical Specification Bases, and ESF Procedure Changes will not result in any changes to the normal operation of the plant. These changes cannot initiate any design basis accidents. Therefore, these changes will not increase the probability or consequences of accidents previously evaluated in the SAR.

UFSAR Table 7-15 and ESF procedure changes will not affect how the Hydrogen Skimmer System performs its design basis function. The system will respond as designed during a design basis accident Large Break LOCA or a high energy line break. The 12 minute time delay assumption for Hydrogen Skimmer System operation in CNC-1552.08-00-0194 will not be impacted by any of these changes. Therefore, the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not increase.

These changes will not adversely impact operation of the Hydrogen Skimmer System as described in the SAR. CNC-1552.08-00-0194, Rev. 3 evaluated hydrogen concentrations within the containment with a Hydrogen Skimmer System 12 minute start delay. This calculation indicated that the Hydrogen Skimmer System is capable of performing its design basis function of maintaining hydrogen concentrations less than 4% volume with a 12 minute time delay and that the Containment Air Return and Hydrogen Skimmer System initiation time is not considered a controlling parameter in the analysis. Therefore, the consequences of an accident previously evaluated in the SAR will not increase.

These changes will not cause any Hydrogen Skimmer System equipment to malfunction. The Hydrogen Skimmer System will operate as designed during a design basis accident. The calculation described above indicated that the system is capable of performing its design basis function of maintaining hydrogen concentrations less than 4% volume with a 12 minute time delay and that the Containment Air Return and Hydrogen Skimmer System initiation time is not considered a controlling parameter in the analysis. Therefore, the consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not increase.

Calculation CNC-1552.08-00-0194 indicated that the Hydrogen Skimmer System is capable of performing its design basis function of maintaining hydrogen concentrations less than 4% by volume with a 12 minute time delay and that the Containment Air Return and Hydrogen Skimmer System System initiation time is not considered a controlling parameter in the analysis. Therefore, the possibility for an accident of a different type than any evaluated previously in the SAR will not be created.

These changes will not affect the operation of any Hydrogen Skimmer System equipment or cause any system equipment to malfunction. The system will operate as designed during a design basis accident. Therefore, the possibility for a different type of malfunction of equipment important to safety than any evaluated previously in the SAR will not be created.

Margin of safety as defined in the Bases for Technical Specifications 3.3.2 and 3.6.8 will not be reduced because CNC-1552.08-00-0194 evaluated performance of the Hydrogen Skimmer System with a 12 minute time delay and determined that hydrogen accumulation would not exceed 4% by volume with the additional time delay. These changes will not compromise performance of the system during a design basis accident. No other Technical Specification Bases are affected. Therefore, the margin of safety as defined in the basis for any Technical Specification will not be reduced.

There are no unreviewed safety questions associated with the changes being made by modification CE-10338, Technical Specification 3.6.8 Bases, or the Hydrogen Skimmer System suction valve ESF Response Times in PT/1(2)/A/4200/09. A Technical Specification Bases change is necessary, but no Technical Specification changes are needed. UFSAR Table 7-15 will be revised.

97 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-10365

Description: Minor Modification CE-10365 replaces instrument 1VXPS5100 and 1VXPS5110 with new models and adds mounting requirements to the associated manual. The Containment Air Return and Hydrogen Skimmer System has two pressure switches (VXPS5100 and VXPS5110) that measure the differential pressure across the pressure boundary between upper and lower containment. Contact outputs from the switches are used to provide interlocks for preventing Containment Air Return Isolation Dampers ARF-D-2 (Train A) and ARF-D-4 (Train B) from opening when there is a high differential pressure across the dampers, thus preventing possible overloading to the damper actuator.

There has been a manufacturer part number change for these pressure switches. The new part is an acceptable substitute for the installed part. It was discovered during a 10-year Environmental Qualification change-out for the Unit 1 switches that there were no documented mounting requirements for the switches. Modification CE-10365 will replace the Unit 1 pressure switches with the new model and mounting requirements will be added to the vendor manual, CNM-1211.00-0446, which is applicable to both Unit 1 and Unit 2.

Evaluation: The function of the Containment Air Return and Hydrogen Skimmer System pressure switches are mentioned in the UFSAR Section 6.2.1.1.3.1 stating that "the damper is prohibited from opening until such time as the pressure differential between the upper and lower compartments is less than 0.5 psi with the lower compartment positive to the upper compartment. The pressure differential permissive is accomplished through a differential pressure switch with normally closed contacts located in the damper motor start circuit. The qualification method and test reference for Solon differential switches, model 7PS11DW, is mentioned in UFSAR Table 3-105 and 3-106, Electrical Equipment Seismic Qualification for Catawba Unit 1 and 2, respectively. The new Solon part number, 7PSW11D2, will be added to these two tables. "Solon 7PS11DW" is also documented in Table 3-3 of Supplement 3 to the SER, NUREG-0954 CNS SER, but is this is not a revisable document. Replacing the Containment Air Return and Hydrogen Skimmer System pressure switches with new switches that are identical in fit, form, and function is essentially a maintenance activity that meets the definition of an "equivalent component" given in Duke Power Nuclear System Directives. There is no unreviewed safety question associated with this modification since the replacement part is considered equivalent to the item it will replace. Therefore the modification will not affect the probability or consequences of accidents analyzed in the UFSAR and no new accident scenarios will be created. No Technical Specification revisions will be necessary. UFSAR Tables 3-105 and 3-106 will be revised to document the environmental qualification of this new part number.

182 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-10367, Revise Containment Process Penetration Design Basis Document.

Description: Minor Modification CE-10367 will add a section to the Containment Design Basis Document (DBD) that will address when to apply the appropriate Technical Specification for leaking capped vents and drains (TVD) within the boundary of Containment Isolation Valves (CIV) and define the Containment Isolation Valve Allowable Stroke Time Bases.

Evaluation: Leaking capped vents and drains (TVD).

The following clarifies leakage from inside and outside TVD with respect to the applicable Technical Specification. Per the bases for Technical Specification 3.6.3, in the event one containment isolation valve in one or more penetration flow paths is inoperable except for purge valve or reactor building bypass leakage not within limit, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve inside containment with flow through the valve secured. For a penetration flow path isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within four hours. The four hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment operability during Modes 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to operable status within the four hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification, through a system walkdown or computer status indication, that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering Mode 4 from Mode 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is unlikely.

A leaking TVD on a containment penetration, inside or outside, has the operator enter Technical Specification 3.6.3. The entry allows a four hour time to isolate or stop the leakage.

The Technical Specification entered is 3.6.3 for CIV's. In the event the CIV's leakage results in exceeding the overall containment leakage rate, (La), Technical Specification

3.6.3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

A leaking TVD on a containment penetration, inside or outside containment, between the CIV's, has the operator enter Technical Specification 3.6.3 condition A. This condition is the same required as if one CIV in the penetration is inoperable. Technical Specifications recognize that the probability of another failed CIV or valve is low, as indicated by the amount of time allowed to isolate this valve, four hours. Since it is assumed that there is no difference between the TVD or the CIV, (just that the valve is inoperable). The TVD leaking inside or outside allows the operator four hours to complete the required action of isolating the penetration or stopping the leak.

CIV Allowable Stroke Times

The bases for the CIV allowable stroke times is not referenced in accessible documents. It should be added to the Containment DBD. This addition to the DBD is for clarification. The bases for the CIV allowable stroke times is taken from ANS 56.2-1984. Reg Guide 1.141 endorses this Standard and it is referenced in UFSAR section 6.2.4.2.1. The closure speed is typically based on the valve size, the requirements for accident radiation dose and emergency core cooling effectiveness. The 60 seconds for closing up containment is based on the largest valve as the slowest operating valve. Other special cases may also require valve closure times different than specified. In determining appropriate valve closure times, consideration shall be given to time delays due to instrument and control delay times as well as valve motive power delay times (diesel start delay time). The Diesel Generator Load Sequencer times (ESF) are excluded here, these are only the CIV allowable stroke times.

Neither of these changes has any effect on the probability or consequences of accidents analyzed in the UFSAR. No new accident scenarios are introduced by these changes. These changes are technical clarifications. The containment isolation criterion addressed in the SAR is not affected. There are no unreviewed safety questions associated with these additions to the Containment DBD. No changes to the UFSAR are required. No Technical Specification changes are required.

187 Type: Minor Modification **Unit:** 1

Title: Minor Modification CE-10390, Correct the setpoint of relief valve 1AS43

Description: Minor Modification CE-10390 will correct the set point of relief valve 1AS43 as shown on flow diagram CN-1595-1.0 Revision 9 from 65 psig to the actual setpoint of 75 psig. The design pressure of the piping upstream of relief valve 1AS43 will be changed to 90 psia to accommodate the 1AS43 set pressure. These changes have been evaluated through the Corrective Action Program. The AS System is the Plant Auxiliary Steam System. Valve 1AS43 is the "Auxiliary Steam System to Auxiliary Building Equipment Relief Valve".

Evaluation: These changes have been evaluated by Engineering. It was determined that there was a small increase in piping stress but that the increase was not significant. There are no unreviewed safety questions associated with this modification. The probability or consequences of accidents evaluated in the UFSAR are not increased. No new accident scenarios are created. No Technical Specifications changes are required. No UFSAR changes are required.

154 Type: Minor Modification **Unit:** 0

Title: Minor Modification CE-10403, Add a note to Low Pressure Service Water System flow diagram and clarify the Power Piping Specification Exclusion Statements

Description: A problem was identified in that there is a discrepancy between the Conventional Low Pressure Service Water System Flow Diagram (CN-1575-1.0) and the Power Piping Specification CNS-1206.00-02-1002. The flow diagram shows 42 inch, 48 inch, and 54 inch piping defined on the flow diagram as Power Piping Specification 150.4. However the Power Piping Specification 150.4 only addresses pipe sizes up to 42 inches. To resolve this discrepancy, this modification requires a note to be added to line listings 01 and 03 on the Conventional Low Pressure Service Water System flow diagram CN-1575-1.0. This modification is also revising Power Piping Specification CNS-1206.00-02-1002 Section 7.3 exclusion statements for both the Conventional Low Pressure Service Water System and the Condenser Circulating Water System to more clearly define the piping specifications that were used during construction and are to be used in the future. The required note should have been included on this flow diagram from the beginning. Neither the Conventional Low Pressure Service Water System nor the Condenser Circulating Water System are nuclear safety related. Neither of these systems' design function or capabilities are being impacted. The changes clarify what specifications are applicable for the piping addressed on the flow diagrams.

Evaluation: This modification will not require any field work. No physical changes will be made to plant structures, systems or components. This modification only makes changes to documentation for clarification. Therefore there will be no effect on the probability or consequences of accidents described in the UFSAR. There are no unreviewed safety questions associated with this modification. No technical specification changes are required. Changes are required for UFSAR Figure 9-63 (piping flow drawing).

211 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-10405, Add new 7300 card instruction books to the vendor manual

Description: Minor Modification CE-10405 will add new 7300 card instruction books to the vendor manual. These instruction books are updates to instruction books currently found in the vendor manuals.

Evaluation: There are no unreviewed safety questions associated with this modification. The function of the circuit cards will not be changed. The new vendor information is a supplement to the previous vendor manual. No Technical Specification changes are required. No UFSAR changes are required.

224 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-10426, Replace 1CMFE6720, 1CMFE6740, 1CMFE6760 with 0.11 orifices

Description: Flush water for the hotwell pump seals is provided from the hotwell pump discharge header. The one half inch flush supply line typically consists of an isolation valve, a flow restricting orifice, a check valve and a 0 to 5 gpm flow meter. The seal requires 1 to 2 gpm seal water per the pump manufacturer. Experience has shown that more than 2 gpm is needed to prevent air inleakage through the pump seals. The existing 0.08 inch orifice is too small to pass the required flow and demineralized water is being used for seal water. This results in a 1 ppb increase in condensate dissolved oxygen. This modification will change the orifice associated with 1CMFE6720, 1CMFE6740, 1CMFE6760 from 0.08 inches to 0.11 inches.

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

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

Title: Minor Modification CE-10428, Replace 2CMFE6720 with 0.11 orifice

Description: Flush water for the hotwell pump seals is provided from the hotwell pump discharge header. The one half inch flush supply line typically consists of an isolation valve, a flow restricting orifice, a check valve and a 0 to 5 gpm flow meter. The seal requires 1 to 2 gpm seal water per the pump manufacturer. Experience has shown that more than 2 gpm is needed to prevent air inleakage through the pump seals. The existing 0.08 inch orifice is too small to pass the required flow and demineralized water is being used for seal water. This results in a 1 ppb increase in condensate dissolved oxygen. This modification will change the orifice associated with 2CMFE6720 from 0.08 inches to 0.11 inches.

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

143 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-10434 , Update UFSAR, Design Basis Documents and System Flow Diagrams to recognize nitrogen as a cover gas for the Reactor Coolant Drain Tank

Description: Minor Modification CE-10434 will add text to UFSAR Sections 11.2.2.1.1, 11.2.2.7.1.1, Table 11-8 and Figure 11-11 as well as to the Liquid Waste Recycle System Design Basis Document to recognize the use of nitrogen, in addition to hydrogen, as a separate cover gas for the Unit 1 and Unit 2 Reactor Coolant Drain Tanks. System Flow diagrams will be updated as well such that either hydrogen or nitrogen can be aligned to the Reactor Coolant Drain Tank gas spaces. A change will be made to the Hydrogen Bulk Storage System flow diagram such that either hydrogen or nitrogen can be aligned to the Reactor Coolant Drain Tank gas spaces. Currently, these documents reflect only the use of hydrogen as a cover gas for the Reactor Coolant Drain Tanks.

In the Reactor Coolant Drain Tanks, hydrogen and fission gases dissolved in reactor coolant come out of solution. Using hydrogen as the Reactor Coolant Drain Tank cover gas provides a transport mechanism to move the fission gases that accumulate in the tank to the Waste Gas System. The hydrogen volume is eliminated in Waste Gas System Recombiners by combining the hydrogen with oxygen to form water. A nitrogen cover gas in the Reactor Coolant Drain Tanks also provides a transport mechanism to the Waste Gas System for the fission and hydrogen gases that accumulate in the tank. In general, a cover gas overpressure of either hydrogen or nitrogen should be kept on the Reactor Coolant Drain Tanks to prevent the ingress of atmospheric oxygen into the system and the possible formation of flammable or explosive gas mixtures. The current practice of using nitrogen as the cover gas during outages prevents the formation of flammable or explosive gas mixtures due to atmospheric oxygen being introduced into the system during maintenance activities.

Evaluation: Using nitrogen as the Reactor Coolant Drain Tank cover gas will not affect the on line functions of the Reactor Coolant Drain Tanks Subsystem in any way. The nitrogen blanket in the tank will be maintained between 2 and 5 psig as currently designed for a hydrogen blanket. The nitrogen cover gas will not affect the ability to use the Reactor Coolant Drain Tanks to measure identified Reactor Coolant System leakage as required by Technical Specification 3.4.13. The presence of the nitrogen blanket will not affect the Reactor Coolant Drain Tanks function of collecting leakage from reactor coolant pump seals, valve stem leakoffs, reactor vessel o-ring or other sources. Because the Reactor Coolant Drain Tanks operating pressure range will not change, the cover gas change will have no effect on reactor coolant pump seals or seal standpipe levels. In addition, the cover gas change will not affect the Reactor Coolant Drain Tanks function of accepting excess letdown heat exchanger effluent generated during startups. Also, using nitrogen as the Reactor Coolant Drain Tank cover gas will not affect the function of valves WL805A and WL807B to provide containment isolation as required by Technical Specification 3.6.3. Finally, a nitrogen blanket on the Reactor Coolant Drain Tanks will not prevent the subsystem from pumping excess leakage to either the Recycle Holdup Tanks or Refueling Water Storage Tanks.

Similarly, replacing the hydrogen cover gas with nitrogen will not affect the refueling functions of the Reactor Coolant Drain Tanks subsystem. The cover gas change will not affect the Reactor Coolant Drain Tanks Subsystem function of: draining the Reactor

Coolant System loops in less than eight hours, recirculating and emptying refueling canal water though the Fuel Pool Cooling demineralizers and filters, and recirculating and emptying refueling canal water through the Boron Recycle demineralizers and filters. Also, a nitrogen cover gas on the Reactor Coolant Drain Tanks will not affect the system's ability to accept drainage form the cold leg accumulators. Finally, a nitrogen blanket on the Reactor Coolant Drain Tanks will not prevent the tank from pumping excess leakage to the Waste Evaporator Feed Tank.

Using nitrogen as the Reactor Coolant Drain Tank cover gas will not appreciably affect the chemistry of the water in the Reactor Coolant Drain Tanks. Reactor Coolant Drain Tanks water that is discharged to the Recycle Holdup Tanks for recycling will eventually pass through the Boron Recycle Evaporator Stripping Column which will remove dissolved nitrogen and other gases to acceptable levels. Reactor Coolant Drain Tank water discharged to the Waste Evaporator Feed Tank is processed as radwaste. As a result, the amount of dissolved gases in the Waste Evaporator Feed Tank water is not a concern.

Replacing the hydrogen Reactor Coolant Drain Tank cover gas with nitrogen will have an insignificant effect on the operation of the Waste Gas System. The effect will be a slight increase in the waste gas inventory. Because nitrogen vented from the Reactor Coolant Drain Tanks will not be recombined in the Waste Gas System hydrogen recombiners, the nitrogen content of the waste gas decay tanks will increase to some extent. The increase will be insignificant due to the small size of the Waste Evaporator Feed Tanks (350 gallons - half of which normally contains liquid) and the infrequent vents to the Waste Gas system. Nitrogen is the primary constituent of the gas circulating through the waste gas loop. The added nitrogen from the Waste Evaporator Feed Tank vents will not affect the Waste Gas systems ability to process, store and release waste gas. In addition, the added nitrogen will not affect the Waste Gas System's ability to monitor and control flammable oxygen/hydrogen gas mixtures as required by Selected Licensee Commitments 16.11-18 and 16.11-20. Also, the added nitrogen will not affect Chemistry's ability to ensure that the curie content of the waste gas decay tanks do not exceed the limits of Selected Licensee Commitment 16.11-19. Finally, using a nitrogen cover gas in the Reactor Coolant Drain Tanks is not considered to be a major change to a gaseous radwaste treatment system, which would be required to be reported to the NRC per SLC 16-11-21.

There are no unreviewed safety questions associated with this minor modification. No Technical Specification changes are required. Changes are required for UFSAR Sections 11.2.2.1.1, 11.2.2.7.1.1, UFSAR Table 11-8 and UFSAR Figure 11-11. Several flow diagram changes will be made as well.

144 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-10461, Revise Diesel Generator Manufacturers Drawing to clarify the required length of the crankcase capscrews

Description: The crankcase doors on the Diesel Engine are connected with 1 - 1/2" and 1 - 1/4", 1/2"-13 capscrews. The Diesel Engine crankcase doors are removed every refueling cycle to perform Improved Technical Specification required inspections. Over the years, the threads in the engine block holes have worn to where the capscrew length listed on the D/G I/M will not hold torque. A longer capscrew is required to hold the required torque. The note being added per this Minor Modification will allow Maintenance to measure the depth of the crankcase door capscrew hole in the engine block and then determine the maximum capscrew length that can be used. This will allow the required torque to be obtained. A note is being added to the respective drawing to give instructions on how to determine the required capscrew length.

Evaluation: The Diesel Engines are designed with crankcase doors to allow internal engine inspections. These doors are removed every outage to perform engine inspections. The crankcase doors are connected to the engine block with with 1 - 1/2" and 1 - 1/4", 1/2"-13 capscrews. A note is being added to the manufacturing drawing to include instructions for Maintenance on how to determine the required capscrew length to ensure the hole depth in the engine block is not exceeded. The capscrews are torqued to 30 - 33 ft--lbs per CNM-1301.00-0237 volume 1 section 8. In order to obtain this required torqued, longer capscrews are required at some locations due to the thread condition of the engine block holes. Allowing an increased capscrew length does not create any concerns as long as the capscrews do not bottom out. Capscrews length is based on a minimum thread engagement, which is not being decreased. Increasing thread engagement is acceptable and does not create any structural problems. The crankcase doors will continue to be connected to the engine block as designed and thus will continue to support operability of the Emergency Diesel Generators.

The Emergency Diesel Generators will continue to be able to perform the Improved Technical Specification requirement of starting, loading, and providing emergency AC power for seven days in response to a Design Basis Accident with the longer capscrews on the engine crankcase doors. The change allowed per this Minor Modification only adds clarification for determining capscrew length. The same number of fasteners will continue to be used to connect the crankcase doors to the engine block. There are no unreviewed safety questions associated with this modification. No UFSAR Changes are required. No Technical Specification Changes are required.

227 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-10528, Control Room Chilled Water System Condenser Waterbox divider plate coatings

Description: Minor Modification CE-10528 addresses actions to restore corroded Control Room Chilled Water System Condenser Waterboxes. This includes application of a protective coating designed for underwater service and a change in bolting material for clamp assembly of Inlet/Outlet waterbox divider plate from carbon steel to stainless steel. The original unprotected carbon steel waterboxes of the Control Room Chilled Water System Condenser have been damaged due to contact with raw lake water from the Nuclear Service Water System which is used for condenser cooling.

Evaluation: There are no unreviewed safety questions associated with this modification. Loss of cooling to the Control Room Chilled Water System Condenser is not identified as an accident initiator. Therefore this modification will not increase the prabability of an accident. The coating material has been evaluated and will provide excellent protection of the carbon steel. The bolting material change was evaluated and determined not to affect the safety function of the system. No Technical Specification changes are required. No UFSAR changes are required.

189 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-10554, Locate Viewports on Standby Shutdown Facility Drawings

Description: Viewports were previously placed in the Standby Shutdown Facility (SSF) per a Work Order. Later it was realized that the modification process should have been used for this work. The walls of the SSF are not nuclear safety related and do not perform a nuclear safety related function. There are certain security scenarios for which the SSF walls provide some degree of protection for SSF equipment and personnel. The addition of the viewports was evaluated for its effect on the security function of the walls.

Evaluation: This modification has no effect on the probability or consequences of accidents evaluated in the UFSAR. There are no Unreviewed Safety Questions associated with this activity. No Technical Specification changes are required. No UFSAR changes are required.

222 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-10706, Delete UFSAR Table 12-28 statements about relative humidity

Description: The Containment Purge Ventilation Systems exhaust lines previously incorporated a high relative humidity trip feature such that a relative humidity of 70% or greater in the exhaust air would automatically close all Containment Purge Ventilation System isolation valves . This feature employed two humidistats per unit. The purpose of the high relative humidity trip was to protect the cleanup filters from long term exposure to humid air. However containment isolation upon high relative humidity was no longer required after the adoption of carbon testing methods at conditions in accordance with ASTM D3803-89. The humidistats were abandoned in place per Modifications CE-61023 and CE-61024. All references to the humidistats and containment isolation on high relative humidity, in the Containment Purge Ventilation System Design Basis Document and the UFSAR were thought to have been removed by these two modifications. A recent review has noted that some references to high humidity still remain. This modification will remove those references.

Evaluation: There are no unreviewed safety questions associated with this modification. No modifications are being made to plant systems, structures, or components. There is no effect on any accident evaluated in the UFSAR and no new accident scenarios are created. No Technical Specification changes are required. UFSAR Table 12-28 will be revised.

230 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-10751, Minor changes to Breathing Air System Flow Diagrams

Description: Breathing Air System Flow Diagrams will be revised to:

1. Identify the existing boundary valves of the receiver tanks.
2. Label the boundary valves
3. Add notes to permit the optional use of the boundary valves for installing level gauge equipment.
4. Correct minor editorial errors currently on the drawings.
5. Improve the equipment identification in support of the Equipment Database and Work Management Systems.

Reclassification of the level gauges on the Breathing Air System receiver tanks as boundary valves and making the glass level gauge optional, does not reduce the effectiveness of the Breathing Air System. The glass gauge was provided as a fluid level indicator. Prior to startup and while in service the Breathing Air System is routinely sampled for breathing air quality with more sophisticated sampling techniques. The glass gauge did not provide a useful service in the breathing air application. The glass gauge was determined a potential system operation and personnel hazard. Therefore the glass was permanently removed. The function will be maintained as optional. The Breathing Air System will continue to operate satisfactorily without the glass gauge.

Evaluation: No unreviewed safety questions are introduced by this change. The probability or consequences of accidents evaluated in the UFSAR are not increased by this change. The Breathing Air System is not an accident initiating system or an accident mitigating system. No Technical Specification changes are required. UFSAR Figures 9-75, 9-76, and 9-77 will be revised.

250 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-10788, Install a replacement fuse holder for device MD46 on 2MC1

Description: Minor Modification CE-10788 replaced an obsolete fuse holder (Bussman Part Number 4575) with the replacement model recommended by the manufacturer (Bussman Part Number BM6032SQ) in Main Control Board 2MC 1. This fuse holder transfers AC power from terminal board MD6 to the 48 VDC power supply MD4. The Digital Rod Position Indication System does not serve a nuclear safety related function therefore the design criteria for single failure, fire protection/appendix R, seismic, electrical separation, equipment qualification, flood, loss of off site power, tornado/wind, missiles and pipe rupture are not applicable. The existing mounting for the fuse holder will also be revised because of the different configuration of the two models. The mounting is seismic related the issue has been evaluated.

Evaluation: Replacement of an obsolete fuse holder with an available replacement recommended by the manufacturer is not an unreviewed safety question. No Technical Specification changes are required. No UFSAR changes are required.

220 Type: Minor Modification **Unit:** 1

Title: Minor Modification CE-10792, Install a one inch drain assembly downstream of Valve 1BB27

Description: Minor Modification CE-10792 will install a one inch drain assembly downstream of Valve 1BB27 on the Steam Generator Blowdown Tank Vent Line. The twelve inch Steam Generator Blowdown Tank vent piping downstream of control valve 1BB27 is filled with water to within ten feet of the Turbine Building roof. This section of piping and its supports are designed assuming the pipe to be empty. The additional weight of a 107 foot water column has caused the spring support to almost bottom out. The additional weight is also producing excessive loads on the nozzle of the Steam Generator Blowdown Tank. This line cannot be used as a vent while it is filled with water due to water hammer concerns. This modification will add a one inch drain valve assembly to the elbow just downstream of control valve 1BB27. This drain assembly will be used to drain the water from the main line until a continuous drain can be designed and installed. The drain will be installed using a wet tap machine and will consist of

1. A one-inch 300 psi half coupling
2. A one inch schedule 40 pipe nipple
3. A one-inch ball valve
4. A one inch schedule 80 threaded pipe nipple
5. A one inch pipe cap

This piping will have a design temperature of 430 degrees F. and a design pressure of 100 psia.

Evaluation: There are no unreviewed safety questions associated with this modification. There will be no effect on the ability of the Steam Generator Blowdown Recycle System to perform its intended design function. No new accident scenarios are created. No Technical Specification changes are required. UFSAR Figure 10-29 will be revised.

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

Title: Minor Modification CE-10794, Install a one inch drain assembly downstream of Valve 2BB27

Description: Minor Modification CE-10794 will install a one inch drain assembly downstream of Valve 2BB27 on the Steam Generator Blowdown Tank Vent Line. The twelve inch Steam Generator Blowdown Tank vent piping down stream of control valve 2BB27 is filled with water to within ten feet of the Turbine Building roof. This section of piping and its supports are designed assuming the pipe to be empty. The additional weight of a 107 foot water column has caused the spring support to almost bottom out. The additional weight is also producing excessive loads on the nozzle of the Steam Generator Blowdown Tank. This line cannot be used as a vent while it is filled with water due to water hammer concerns. This modification will add a one inch drain valve assembly to the elbow just downstream of control valve 2BB27. This drain assembly will be used to drain the water from the main line until a continuous drain can be designed and installed. The drain will be installed using a wet tap machine and will consist of

1. A one-inch 300 psi half coupling
2. A one inch schedule 40 pipe nipple
3. A one-inch ball valve
4. A one inch schedule 80 threaded pipe nipple
5. A one inch pipe cap

This piping will have a design temperature of 430 degrees F. and a design pressure of 100 psia.

Evaluation: There are no unreviewed safety questions associated with this modification. There will be no effect on the ability of the Steam Generator Blowdown Recycle System to perform its intended design function. No new accident scenarios are created. No Technical Specification changes are required. UFSAR Figure 10-31 will be revised.

258 Type: Minor Modification **Unit:** 0

Title: Minor Modification CE-10850, Revise drawings to show temporary lead shielding as a permanent installation.

Description: Elevated dose rates were experienced originating from the Unit 2 primary sample sink drain lines. In order to reduce exposure to station personnel, temporary lead shielding was installed. Removal of this shielding was pending a permanent solution to the radioactive material buildup in this drain piping. It was also determined the dose rates on the Unit 1 drain lines are higher than on the Unit 2 lines. The Unit 1 piping is covered with temporary lead shielding blankets attached to the pipe. Minor Modification CE-10850 is to authorize the existing shielding installation as permanent.

Evaluation: Both the Nuclear Sampling System and the Liquid Radwaste System portions impacted by this modification are non-nuclear safety related piping and components. Design for seismic loading is not required for this portion of these systems and neither performs a safety related function in mitigating the consequences of accidents. There is no unreviewed safety question associated with this modification. No Technical Specification changes are required. No UFSAR changes are required

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

Title: Minor Modification CE-61291, Remove internals from check valves 1CA171 and CA172

Description: Minor Modification CE-61291 removes the valve internals (disc assembly and associated hardware) from check valves 1CA171 and 1CA172. After this modification these valves will no longer require Inservice Testing and they will be removed from the IST Program. Valves 1CA171 and 1CA172 are in the Nuclear Service Water System supply piping to the Auxiliary Feedwater System Pumps. These valves are no longer required due to the recent installation of valves 1CA291 and 1CA292. Check valves 1CA291 and 1CA292 were installed by minor modification CN-61239 to ensure train separation of the Auxiliary Feedwater System motor driven pumps supplies from the Nuclear Service Water System.

The new check valves perform the functions originally performed by valves 1CA171 and 1CA172. These functions include preventing gross diversion of flow from one Nuclear Service Water train header through a failed header on the opposite Nuclear Service Water train and providing secondary side isolation during an SSS Event. Therefore valves 1CA171 and 1CA172 are no longer needed and their internals can be removed without loss of function. The Auxiliary Feedwater System and its assured source of Nuclear Service Water will still serve their normal operation and accident mitigation functions.

Evaluation: This modification does not involve an unreviewed safety question. The function of the two valves will be assumed by two newly installed valves. No Technical Specification changes are required. UFSAR Figure 10-33 (a piping flow diagram) will be revised to show that the internals of valves 1CA171 and 1CA172 have been removed.

118 Type: Minor Modification **Unit:** 1

Title: Minor Modification CE-61311, Delete valve position transmitters 1CFVP0060 and 1CFVP0130 from valves 1CF6 and 1CF13

Description: Minor Modification CE-61311 will delete valve position transmitters 1CFVP0060 and 1CFVP0130 from valves 1CF6 and 1CF13. The existing transmitters are obsolete and the position signals they provide to the Operator Aid Computer are no longer required. Each transmitter is connected to its respective valve positioner through a mechanical linkage. Electrical connections are provided for 120 VAC Power to the transmitters and for analog signal input to the Operator Aid Computer. This modification will mechanically and electrically disconnect each transmitter and remove it from the local panel. Other position indication for valves 1CF6 and 1CF13 is provided on the Main Control Board.

Evaluation: The Feedwater Pump recirculation valves are positioned by manual loaders to automatically maintain a minimum flow from each feedwater pump by recirculating the pump discharge back to the main condenser. While there are other controls involved in the operation of these valves, the valve position transmitters deleted by the modification have served only to provide position indication on the plant Operator Aid Computer. The recirculation valves and all their associated controls and instrumentation are not nuclear safety related. The valves and their positioners and position transmitters are located in the Turbine Building. Deletion of the position transmitters will have no effect on plant safety or on any accidents evaluated in the UFSAR. There is no unreviewed safety question associated with this modification. No Technical Specification changes are required. A revision will be made to UFSAR Figure 10-27, a system flow diagram.

13 Type: Minor Modification **Unit:** 1

Title: Minor Modification CE-61336, Provide a permanent emergency eyewash/shower facility at the Unit 1A and 1B Containment Spray System Chemical Handling Areas

Description: Minor Modification CE-61336 will provide a permanent emergency eyewash/shower facility at the Unit 1A and 1B Containment Spray System Chemical Handling Areas. This installation will replace a temporary portable emergency eyewash/shower. The purpose of the Containment Spray System Chemical Handling Areas is to provide a means to inject chemicals into the Containment Spray Heat Exchanger wet layup loops in the Nuclear Service Water System. The modification affects piping and equipment located in the Auxiliary Building. The Makeup Demineralized Water System will provide the source of water for the facility.

Evaluation: The Makeup Demineralized Water System is not required for maintenance of plant safety in the event of an accident. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. UFSAR Figure 9-45 (piping flow drawing) will be revised.

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

Title: Minor Modification CE-61337, Provide a permanent emergency eyewash/shower facility at the Unit 2A and 2B Containment Spray System Chemical Handling Areas

Description: Minor Modification CE-61337 will provide a permanent emergency eyewash/shower facility at the Unit 2A and 2B Containment Spray System Chemical Handling Areas. This installation will replace a temporary portable emergency eyewash/shower. The purpose of the Containment Spray System Chemical Handling Areas is to provide a means to inject chemicals into the Containment Spray Heat Exchanger wet layup loops in the Nuclear Service Water System. The modification affects piping and equipment located in the Auxiliary Building. The Makeup Demineralized Water System will provide the source of water for the facility.

Evaluation: The Makeup Demineralized Water System is not required for maintenance of plant safety in the event of an accident. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. UFSAR Figure 9-45 (piping flow drawing) will be revised.

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

Title: Minor Modification CE-61374, Replace Roofs on Various Buildings

Description: Minor Modification CE-61374 replaces the roofs on various station buildings. The modification will allow roofing replacement on several portions of the Auxiliary Building, the entire Standby Shutdown Facility, the Spent Fuel Canopy Buildings, the Lube Oil Storage Building, and the Hydrogen, Nitrogen, and Oxygen Gas Storage Buildings. The old roofing will be removed and new roofs will be installed. The new roofing will consist of base layers of insulation, topped with a smooth modified bituminous membrane and a mineral surfaced sheet. This work will include new flashing and accessories. The work will increase the finished elevation of the roofs in some places. The affected buildings are inside the Protected Area.

Evaluation: This modification does not change the function of the buildings. There will be no effect on the operation of the station after the work is completed. The building roofs are not accident initiators. There is a possibility that the carbon beds of the Auxiliary Building Ventilation System could be affected if solvent fumes from the roofing process were drawn into the system. The work will be closely controlled to ensure that this does not happen. However if it did happen Technical Specification 4.7.7 requires a surveillance to be performed which would prove operability or cause entry into an action statement. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. No UFSAR chnages are required.

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

Title: Minor Modification CE-61391

Description: Minor Modification CE-61391 replaces the four Refueling Water Storage Tank level transmitters with different model transmitters capable of remaining operable when submerged. This is required in case of a Refueling Water Storage Tank rupture.

Evaluation: There are no unreviewed safety questions associated with this minor modification. Failure of any or all of these transmitters by themselves cannot initiate any accident evaluated in the UFSAR. The new transmitters are equal or better in performance compared to the existing transmitters and there are no new failure modes other than those already present with the existing transmitters. The function of the transmitters during accident conditions will not change and the ability of the new transmitters to perform this function will be equal to or better than the existing transmitters. There are no new failure modes created by installation of the new transmitters. The new transmitters meet existing accuracy requirements for the Refueling Water Storage Tank level instrumentation.. Therefore no margins of safety as defined in any Technical Specification will be reduced. No Technical Specification changes are required. No UFSAR changes are required.

172 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-61394, Add time delay to the manual safety injection reset logic

Description: Minor Modification CNCE-61394 modifies the Catawba Unit 1 manual safety injection (SI) reset circuits to include a reset time delay. The purpose of this time delay is to ensure that each of the safety injection slave relays is given a fixed minimum duration of reset voltage (i.e., unlatching voltage) that is independent of the operation of any one of the slave relays to be unlatched.

Operating experience at several utilities has shown that faster than normal operation of the K602 slave relay, or slower than normal operation of the other safety-injection slave relays (K601, K603, K604, K608, K611, K630), can result in incomplete resetting of the safety injection signal. The original circuit design utilizes a contact from K602 to remove the unlatching voltage to all of the other slave relays. When one or more of the slave relays is slow to unlatch, or when K602 operates much faster than the other slave relays, all of the slave relays may not be able to reset.

This modification utilizes a time-delay dropout relay to control removal of the unlatching voltage. When the Manual Safety Injection (SI) Reset pushbutton on the control board is operated, time-delay relay TD2 will maintain its contacts in the energized position for a time delay period set between 0.5 to 5 seconds.

A seismic evaluation of the Solid State Protection System (SSPS) Output Cabinet has been performed to consider the effects of adding an additional relay. The addition of the relay has been found to be acceptable, and the seismic calculation (CNC-1381.05-00-0111) will be revised. The SSPS cabinets are located in a mild environment.

Evaluation: The revised safety injection reset circuit functions essentially the same as in the original design except that an interposing time delay dropout relay connects the reset voltage to the related slave relay unlatching coils. Train separation is maintained in the revised design such that a single failure cannot affect both trains of the SSPS. Any failure of a reset time-delay relay could only affect the train in which it is installed.

Any failure proposed for the new relay would result in consequences that are no different than the potential failures in the existing circuit components. The conceivable failures for the new relays are that the contacts fail to close when SI reset is required; or that the contacts fail to open after the time-delayed reset period. Either of these failures would produce results that have always been a possible result of a failure of either the SI Reset switch or existing circuit relay TDI.

This modification involves no Unreviewed Safety Questions. No changes to the Technical Specifications are required. UFSAR Figure 7-2 page 8 will be revised.

The addition of a time delay dropout relay in the reset circuit for safety injection will not increase the probability of an accident of any kind. Any conceivable failure of the new relay would produce results identical to failures of the existing circuit components.

The probability of an equipment malfunction important to safety, specifically the SSPS, is not increased. The added relays are normally de-energized and function only to reset the

SI slave relays. Failures inherent in the existing circuit components produce the same results as any failure of the new reset relay.

The response of the SSPS to an accident will not be affected by this modification of the SI reset logic. The modification assures that the SI slave relays are provided a reset signal for a period long enough to ensure that they all can unlatch. The consequences of accidents previously evaluated in the UFSAR are not affected.

The consequences of a malfunction of the SI reset circuit are not altered by this modification. The potential for incomplete resetting of SI is reduced by providing an assured duration unlatch voltage to all of the related slave relays. This modification will not affect the consequences of an equipment malfunction that has been previously evaluated in the UFSAR.

The SI reset circuit changes made by this modification will be virtually transparent to system operation. The SI reset signal will be maintained for five seconds or less. This modification does not create the possibility for any type of accident not previously evaluated in the UFSAR.

The changes made by this modification will not result in any other malfunction of safety related equipment that has not been evaluated in the UFSAR. The conceivable failures for the SI reset circuit will have the same results after the modification as in the original design. The existing potential for incomplete resetting of an SI (which may not have been previously analyzed) is minimized by the addition of the reset time delay.

The SI reset function is not discussed in the Technical Specifications, as such, the margins of safety defined in the Technical Specifications will not be reduced by this modification.

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

Title: Minor Modification CE-61395, Add time delay to the manual safety injection reset logic

Description: Minor Modification CNCE-61395 modifies the Catawba Unit 2 manual safety injection (SI) reset circuits to include a reset time delay. The purpose of this time delay is to ensure that each of the safety injection slave relays is given a fixed minimum duration of reset voltage (i.e., unlatching voltage) that is independent of the operation of any one of the slave relays to be unlatched.

Operating experience at several utilities has shown that faster than normal operation of the K602 slave relay, or slower than normal operation of the other safety-injection slave relays (K601, K603, K604, K608, K611, K630), can result in incomplete resetting of the safety injection signal. The original circuit design utilizes a contact from K602 to remove the unlatching voltage to all of the other slave relays. When one or more of the slave relays is slow to unlatch, or when K602 operates much faster than the other slave relays, all of the slave relays may not be able to reset.

This modification utilizes a time-delay dropout relay to control removal of the unlatching voltage. When the Manual Safety Injection (SI) Reset pushbutton on the control board is operated, time-delay relay TD2 will maintain its contacts in the energized position for a time delay period set between 0.5 to 5 seconds.

A seismic evaluation of the Solid Stae Protection System (SSPS) Output Cabinet has been performed to consider the effects of adding an additional relay. The addition of the relay has been found to be acceptable, and the seismic calculation (CNC-1381.05-00-0111) will be revised. The SSPS cabinets are located in a mild environment.

Evaluation: The revised safety injection reset circuit functions essentially the same as in the original design except that an interposing time delay dropout relay connects the reset voltage to the related slave relay unlatching coils. Train separation is maintained in the revised design such that a single failure cannot affect both trains of the SSPS. Any failure of a reset time-delay relay could only affect the train in which it is installed.

Any failure proposed for the new relay would result in consequences that are no different than the potential failures in the existing circuit components. The conceivable failures for the new relays are that the contacts fail to close when SI reset is required; or that the contacts fail to open after the time-delayed reset period. Either of these failures would produce results that have always been a possible result of a failure of either the SI Reset switch or existing circuit relay TDI.

This modification involves no Unreviewed Safety Questions. No changes to the Technical Specifications are required. UFSAR Figure 7-2 page 8 will be revised.

The addition of a time delay dropout relay in the reset circuit for safety injection will not increase the probability of an accident of any kind. Any conceivable failure of the new relay would produce results identical to failures of the existing circuit components.

The probability of an equipment malfunction important to safety, specifically the SSPS, is not increased. The added relays are normally de-energized and function only to reset the

SI slave relays. Failures inherent in the existing circuit components produce the same results as any failure of the new reset relay.

The response of the SSPS to an accident will not be affected by this modification of the SI reset logic. The modification assures that the SI slave relays are provided a reset signal for a period long enough to ensure that they all can unlatch. The consequences of accidents previously evaluated in the UFSAR are not affected.

The consequences of a malfunction of the SI reset circuit are not altered by this modification. The potential for incomplete resetting of SI is reduced by providing an assured duration unlatch voltage to all of the related slave relays. This modification will not affect the consequences of an equipment malfunction that has been previously evaluated in the UFSAR.

The SI reset circuit changes made by this modification will be virtually transparent to system operation. The SI reset signal will be maintained for five seconds or less. This modification does not create the possibility for any type of accident not previously evaluated in the UFSAR.

The changes made by this modification will not result in any other malfunction of safety related equipment that has not been evaluated in the UFSAR. The conceivable failures for the SI reset circuit will have the same results after the modification as in the original design. The existing potential for incomplete resetting of an SI (which may not have been previously analyzed) is minimized by the addition of the reset time delay.

The SI reset function is not discussed in the Technical Specifications, as such, the margins of safety defined in the Technical Specifications will not be reduced by this modification.

16 Type: Minor Modification **Unit:** 0

Title: Minor Modification CE-61404, Provide permanent Eyewash/Shower facilities at the Control Room Chiller Chemical Handling Area

Description: Minor Modification CE-61404 provides permanent Eyewash/Shower facilities at the Control Room Chiller Chemical Handling Area. This installation will replace a temporary portable emergency eyewash/shower. The modification affects piping and equipment located in the Auxiliary Building. The Makeup Demineralized Water System will provide the source of water for the facility.

Evaluation: The Makeup Demineralized Water System is not required for maintenance of plant safety in the event of an accident. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. UFSAR Figure 9-45 (piping flow drawing) will be revised.

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

Title: Minor Modification CE-61436, Replace vacuum switches on certain radiation monitors with mass flow computers

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

EMF 41 Auxiliary Building Single Range Beta Monitor
EMF43A Control Room Air Intake Monitor
EMF43B Control Room Air Intake Monitor
1EMF33 Condenser Air Ejector Exhaust Monitor
1EMF42 Fuel Building Ventilation Monitor

On each of these non nuclear safety related radiation monitors, the existing vendor supplied vacuum switch used for high and low flow alarms and pump control interlocks is being 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 being mounted on the radiation monitor skid and connected to provide all of the functions of the vacuum instruments and rotameter. This modification was caused by previously encountered problems with the vacuum based flow instruments.

Evaluation: All of the radiation monitors affected by the modification are not nuclear safety related. Both the old and the replacement radiation monitor flow instrumentation are non-nuclear safety related. The new instrumentation installed by this modification performs the same system function as did the instrumentation they replaced. Any failure of the new instrumentation would have the identical effect on the operation of their related radiation monitors as failures of the old instrumentation. However the new instrumentation is expected to provide much improved accuracy and reliability. This modification will have no effect on the probability or consequences of accidents described in the UFSAR. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. UFSAR Figures 9-108, 9-118, 9-122 and 10-13 (piping flow drawings) will be revised.

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

Title: Minor Modification CE-61437, Replace vacuum switches on certain radiation monitors with mass flow computers

Description: Minor Modification CE-61437 changes the sample flow instrumentation on the following Unit 2 Process Radiation Monitors:
2EMF33 Condenser Air Ejector Exhaust Monitor
2EMF42 Fuel Building Ventilation Monitor
On each of these non nuclear safety related radiation monitors, the existing vendor supplied vacuum switch used for high and low flow alarms and pump control interlocks is being 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 being mounted on the radiation monitor skid and connected to provide all of the functions of the vacuum instruments and rotameter. This modification was caused by previously encountered problems with the vacuum based flow instruments.

Evaluation: All of the radiation monitors affected by the modification are not nuclear safety related. Both the old and the replacement radiation monitor flow instrumentation are non-nuclear safety related. The new instrumentation installed by this modification performs the same system function as did the instrumentation they replaced. Any failure of the new instrumentation would have the identical effect on the operation of their related radiation monitors as failures of the old instrumentation. However the new instrumentation is expected to provide much improved accuracy and reliability. This modification will have no effect on the probability or consequences of accidents described in the UFSAR. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. UFSAR Figure 10-13 (piping flow drawing) will be revised.

17 Type: Minor Modification **Unit:** 1

Title: Minor Modification CE-61452, Revise Diesel Generator Lube Oil Sump Tank Level Instrumentation Setpoint

Description: Minor Modification CE-61452 revises the Diesel Generator Lube Oil Sump Tank Level Instrumentation Setpoint. The Improved Technical Specifications have a new surveillance requirement (3.8.3.2) to verify lube oil sump tank level is greater than 400 gallons. The Diesel Generator Lube Oil Tank Level alarm will be increased to 415 gallons to ensure that Operations is aware of a decreasing oil level prior to failing a surveillance requirement. Currently this alarm is set at 385 gallons.

Evaluation: The diesel generators are not accident initiators. This modification changes an alarm setoint to be compatible with the Improved Technical Specification. The required volume of lube oil to support Emergency Diesel Generator operation is not being changed nor is the normal operating system volume. There is no unreviewed safety question associated with this modification. No Technical Specification changes are required. UFSAR Section 9.5.7.2.1 will be revised.

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

Title: Minor Modification CE-61453, Revise Diesel Generator Lube Oil Sump Tank Level Instrumentation Setpoint

Description: Minor Modification CE-61453 revises the Diesel Generator Lube Oil Sump Tank Level Instrumentation Setpoint. The Improved Technical Specifications have a new surveillance requirement (3.8.3.2) to verify lube oil sump tank level is greater than 400 gallons. The Diesel Generator Lube Oil Tank Level alarm will be increased to 415 gallons to ensure that Operations is aware of a decreasing oil level prior to failing a surveillance requirement. Currently this alarm is set at 385 gallons.

Evaluation: The diesel generators are not accident initiators. This modification changes an alarm setoint to be compatible with the Improved Technical Specification. The required volume of lube oil to support Emergency Diesel Generator operation is not being changed nor is the normal operating system volume. There is no unreviewed safety question associated with this modification. No Technical Specification changes are required. UFSAR Section 9.5.7.2.1 will be revised.

19 Type: Minor Modification **Unit:** 0

Title: Minor Modification CE-61455, Relocation and Update of Technical Specification Interpretation Information to Design Basis Documents for various systems

Description: Minor Modification CE-61455 will relocate and update Technical Specification Interpretation Information to Design Basis Documents for the Auxiliary Feedwater System, the Refueling Water System, the Boron Recycle System, the Reactor Coolant System, the Residual Heat Removal System, the Ice Condenser Refrigeration System, the Safety Injection System, the Containment Spray System, the Chemical and Volume Control System and the Containment Penetrations. No hardware changes or procedures are involved with this modification. These are editorial changes associated with retaining or explaining descriptive information concerning important safety system requirements and support functions, and do not alter any safety analysis assumptions, flooding analysis, or other plant design bases. The LTOP discussion added to the Safety Injection System and Chemical and Volume Control System Design Bassis documents do not allow operation of the plant to deviate from the approved requirements of the Technical Specifications or the Selected Licensee Commitments.

Evaluation: There is no change to to any design limit or setpoint. No control function, instrument function, or performance of any structure, system, or component is degraded. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. No UFSAR changes are required.

216 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-61472, Replace vacuum switches on Unit 1 radiation monitors

Description: Minor Modification CE-61472 changes the sample flow instrumentation on the following Unit 1 Process Radiation Monitors:
1EMF35, 1EMF36, 1EMF37 Unit Vent Particulate, Iodine and Gas Monitors
1EMF38, 1EMF39, 1EMF40 Containment Atmosphere Particulate, Iodine and Gas Monitors
On each of these radiation monitors, the existing vendor supplied vacuum switch used for high and low flow alarms and pump control interlocks is being deleted. On 1EMF35, 1EMF36, and 1EMF37 the vacuum switch provided for automatic flow control valve is being deleted since this feature is not used. The existing vacuum gauge and in-line rotameter are also being deleted. A new in-line mass flow element and mass flow computer are being mounted on each of the radiation monitor skids and are connected to provide all of the functions of the vacuum instruments and rotameter. Automatic valve control is included with the new instrumentation.

This modification was caused by problems with the vacuum based flow instruments.

Evaluation: The modification of these radiation monitoring instruments is considered a direct replacement. The functions of the monitors are not being changed and the new monitors have been evaluated to be adequate replacements of the old instruments. There is no unreviewed safety question associated with this modification. There is no effect on any accident evaluated in the UFSAR and no new accident scenarios are created. No Technical Specification changes are required. UFSAR Figure 9-120 will be revised.

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

Title: Minor Modification CE-61473, Replace vacuum switches on Unit 2 radiation monitors

Description: Minor Modification CE-61473 changes the sample flow instrumentation on the following Unit 2 Process Radiation Monitors:
2EMF35, 2EMF36, 2EMF37 Unit Vent Particulate, Iodine and Gas Monitors
2EMF38, 2EMF39, 2EMF40 Containment Atmosphere Particulate, Iodine and Gas Monitors
On each of these radiation monitors, the existing vendor supplied vacuum switch used for high and low flow alarms and pump control interlocks is being deleted. On 2EMF35, 2EMF36, and 2EMF37 the vacuum switch provided for automatic flow control valve is being deleted since this feature is not used. The existing vacuum gauge and in-line rotameter are also being deleted. A new in-line mass flow element and mass flow computer are being mounted on each of the radiation monitor skids and are connected to provide all of the functions of the vacuum instruments and rotameter. Automatic valve control is included with the new instrumentation.

This modification was caused by problems with the vacuum based flow instruments.

Evaluation: The modification of these radiation monitoring instruments is considered a direct replacement. The functions of the monitors are not being changed and the new monitors have been evaluated to be adequate replacements of the old instruments. There is no unreviewed safety question associated with this modification. There is no effect on any accident evaluated in the UFSAR and no new accident scenarios are created. No Technical Specification changes are required. No UFSAR revisions are required.

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

Title: Minor Modification CE-61477, Provide flow measuring instrumentation, an isolation valve and a vent valve in the Nuclear Service Water System Train 1B Supply Piping to the Auxiliary Feedwater System and replace approximately eleven feet of piping.

Description: Minor Modification CE-61477 will provide flow measuring instrumentation, an isolation valve, and a vent valve in the Nuclear Service Water System Train 1B Supply Piping to the Auxiliary Feedwater System and replace approximately eleven feet of the six inch Nuclear Service Water System piping with eight inch piping. Also, an editorial change is being made on flow diagram CN-1604-1.2 to correct a continuation flag error. Providing this instrumentation will allow testing to be performed to determine the piping roughness of the Nuclear Service Water System piping. Replacing the six inch piping with the eight inch piping results in less resistance to flow. An associated variation notice provides Test Acceptance Criteria to be used during flow testing to ensure Nuclear Service Water supply piping to the Auxiliary Feedwater System Pumps provides acceptable flow to meet the Auxiliary Feedwater System design basis.

Evaluation: This modification affects the safety-related assured source Nuclear Service Water System Train 1B supply piping to Auxiliary Feedwater System Pumps - Motor Driven 1B and Turbine Driven No. 1. This piping will be modified to include an isolation valve (1RNE93), a vent valve (1RNE94) and flow instrumentation (1RNFE9290 and 1RNFX9290). Also, a portion of this six inch piping will be replaced with eight inch piping. There are no electrical cables or power requirements associated with this modification.

Variation Notice VN-61477D provides Test Acceptance Criteria to be used during the flow testing of the Nuclear Service Water System supply piping to the Auxiliary Feedwater System pumps. These criteria were developed from calculation CNC-1223.42-00-0001 revision 12B. A new Test Acceptance Criteria sheet was originated to provide acceptable flow test pressure differentials for various flow rates through the six inch Unit 1 Nuclear Service Water Train 1B to Auxiliary Feedwater System piping from the 24 inch header to just upstream of valve 1RN310B. Meeting this test acceptance criteria ensures that Nuclear Service Water System to Auxiliary Feedwater System pumps is sufficient to meet Auxiliary Feedwater System Design Bases. An approved station procedure (PT/1/A/4400/014) will be used to measure and record the pressure differentials.

The design requirements of this portion of the Nuclear Service Water System are Duke Class C (Nuclear Safety Related), carbon steel, 150 psig and 150 degrees F. The new isolation valve (1RNE93) is an eight inch wafer butterfly valve. This valve is a stainless steel nuclear safety related valve with design conditions of 150 psig at 200 degrees F. and will meet the design temperature and pressure requirements of the Nuclear Service Water System as stated above. Stainless steel is an acceptable material for use in this portion of the Nuclear Service Water System. The new vent valve is a one inch Y-type globe valve. This valve is a carbon steel nuclear safety related Class A valve with design conditions of 2735 psig at 680 degrees F. and will meet the design temperature and pressure requirements of the Nuclear Service Water System as stated above. All the new piping and fittings are nuclear safety related carbon steel and will also meet the design requirements. The new globe valve is suitable for use in the Nuclear Service Water System as a vent valve. The new butterfly valve is suitable for use in the Nuclear Service

Water System as an isolation valve. The new instrumentation configuration will include a set of orifice flanges, an orifice plate to be installed during testing, a blank orifice plate to be installed during normal system operation, and the associated instrumentation tubing, root valves and manifold valve. All these instrumentation components are of nuclear safety related material and acceptable for use in this portion of the Nuclear Service Water System. The addition of the weight of the new piping and instrumentation components has been evaluated for impact to the stress analysis, and support/restraint modifications are not required. Adding these valves, piping components and instrumentation components will not affect the operation or function of the Nuclear Service Water or Auxiliary Feedwater Systems during any phase of normal or accident mitigation operation. The Nuclear Service Water and Auxiliary Feedwater systems will continue to function as described in the UFSAR and the design basis specifications.

During normal and emergency operation, the blank flow orifice is installed in the piping. The orifice plate used for testing will only be installed when this portion of the Nuclear Service Water System is out of service for flow testing which will be controlled by an approved station periodic testing (PT) procedure. Auxiliary Feedwater System flow calculations have been revised to evaluate the effect that this modification (with blank orifice plate installed) will have on the Nuclear Service Water flow to the Auxiliary Feedwater System during normal and emergency operation. Based on a calculation to compare the new resistance factor with the existing resistance factor, the impact to the flow rates was found to be negligible.

Modification CE-61477 and associated variation notice VN-61477D do not involve an Unreviewed Safety Question. No changes to the Technical Specifications are required. UFSAR changes are required for Figure 9-31 (the Nuclear Service Water System Flow Diagram CN-1574-2.5) to show the new components added by this modification.

92 Type: Minor Modification **Unit:** 1

Title: Minor Modification CE-61478, Provide flow measuring instrumentation, an isolation valve and a vent valve in the Nuclear Service Water System Train 1A Supply Piping to the Auxiliary Feedwater System and replace a portion of piping.

Description: Minor Modification CE-61478 and Variation Notice VN-61478C will provide flow measuring instrumentation, an isolation valve, and a vent valve in the Nuclear Service Water System Train 1A Supply Piping to the Auxiliary Feedwater System Pumps - Motor Driven 1A and Turbine Driven Number 1. Also a portion of the six inch Nuclear Service Water System piping will be replaced with eight inch piping. Providing this instrumentation will allow testing to be performed to determine the piping roughness of the Nuclear Service Water System piping. Replacing the six inch piping with eight inch piping results in less resistance to flow. VN-61478C provides test acceptance criteria to be used during flow testing to ensure the Nuclear Service Water System supply piping to the Auxiliary Feedwater System pumps provide acceptable flow to meet the Auxiliary Feedwater System design basis.

Evaluation: This modification affects the safety-related assured source Nuclear Service Water System Train 1A supply piping to Auxiliary Feedwater System Pumps - Motor Driven 1A and Turbine Driven No. 1. This piping will be modified to include an isolation valve (1RNE95), a vent valve (1RNE96) and flow instrumentation (1RNFE9280 and 1RNFX9280). Also, a portion of this six inch piping will be replaced with eight inch piping. There are no electrical cables or power requirements associated with this modification.

Variation Notice VN-61478C provides test acceptance criteria to be used during the flow testing of the Nuclear Service Water Supply piping to the Auxiliary Feedwater System Pumps. This information was developed per calculation CNC-1223.42-00-0001, Rev 13A. A new test acceptance criteria sheet was originated to provide acceptable flow test pressure differentials for various flow rates through the six inch Unit 1 Nuclear Service Water System Train 1A to Auxiliary Feedwater piping from the 24 inch header to just upstream of valve 1RN250A. Meeting this criteria will ensure that Nuclear Service Water System flow to the Auxiliary Feedwater System pumps is sufficient to meet the Auxiliary Feedwarer System design basis. An approved station procedure (PT/1/A/4400/014) will be used to measure and record the pressure differentials.

The design requirements of this portion of the Nuclear Service Water System are Duke Class C (Nuclear Safety Related), carbon steel, 150 psig and 150 degrees F. The new isolation valve (1RNE95) is a ten inch wafer butterfly valve. This valve is a stainless steel nuclear safety related valve with design conditions of 150 psig at 200 degrees F. and will meet the design temperature and pressure requirements of the Nuclear Service Water System as stated above. Stainless steel is an acceptable material for use in this portion of the Nuclear Service Water System. The new vent valve is a one inch Y-type globe valve. This valve is a carbon steel nuclear safety related Class A valve with design conditions of 2735 psig at 680 degrees F. and will meet the design temperature and pressure requirements of the Nuclear Service Water System as stated above. All the new piping and fittings are nuclear safety related carbon steel and will also meet the design requirements. The new butterfly valve is suitable for use in the Nuclear Service Water System as an isolation valve. The new instrumentation configuration will include a set of orifice flanges, an orifice plate to be installed during testing, a blank orifice plate to be installed during normal system operation, and the associated instrumentation tubing, root valves and manifold valve. Except for the two orifice plates, all these instrumentaion components are nuclear safety related material and are acceptable for use in this portion of the Nuclear Service Water System. The two orifce plates are of material that is not nuclear safety related because they are not pressure boundary components, however they are acceptable for use in the Nuclear Sarvice Water System. The addition of the weight of the new piping and instrumentation components has been evaluated for impact to the stress analysis, and support/restraint modifications are being modified as required. Adding these valves, piping components and instrumentation components will not affect the operation or function of the Nuclear Service Water or Auxiliary Feedwater Systems during any phase of normal or accident mitigation operation. The Nuclear Service Water and Auxiliary Feedwater systems will continue to function as described in the UFSAR and the design basis specifications.

During normal and emergency operation, the blank flow orifice is installed in the piping. The orifice plate used for testing will only be installed when this portion of the Nuclear Service Water System is out of service for flow testing which will be controlled by an approved station periodic testing (PT) procedure. Auxiliary Feedwater System flow calculations have been revised to evaluate the effect that this modification will have on the Nuclear Service Water flow to the Auxiliary Feedwater System during normal and emergency operation. Based on a calculation to compare the new resistance factor with the existing resistance factor, the impact to the flow rates was found to be negligible.

Modification CE-61478 does not involve an Unreviewed Safety Question. No changes to the Technical Specifications are required. No UFSAR changes are required.

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

Title: Minor Modification CE-61479, Provide flow measuring instrumentation, an isolation valve and a vent valve in the Nuclear Service Water System Train 2A Supply Piping to the Auxiliary Feedwater System and replace a portion of piping.

Description: Minor Modification CE-61479 and Variation Notice 61479E will provide flow measuring instrumentation, an isolation valve, and relocates an existing vent valve in the Nuclear Service Water System Train 2A Supply Piping to the Auxiliary Feedwater System Pumps - Motor Driven 2A and Turbine Driven No. 2. Also, the modification will replace a portion of the six inch Nuclear Service Water System piping with eight inch piping. Providing this instrumentation will allow testing to be performed to determine the piping roughness of the Nuclear Service Water System piping. Replacing the six inch piping with eight inch piping results in less resistance to flow. VN-61479E provides test acceptance criteria to be used during flow testing to ensure the Nuclear Service Water System supply piping to the Auxiliary Feedwater System pumps provide acceptable flow to meet the Auxiliary Feedwater System design basis.

Evaluation: This modification affects the safety-related assured source Nuclear Service Water System Train 2A supply piping to Auxiliary Feedwater System Pumps - Motor Driven 2A and Turbine Driven No. 2. This piping will be modified to include an isolation valve (2RNE95) and flow instrumentation (2RNFE9280 and 2RNFX9280). Existing vent valve 2RNC34 will be relocated on the six inch piping. Also, a portion of this six inch piping will be replaced with eight inch piping. There are no electrical cables or power requirements associated with this modification.

Variation Notice VN-61478E provides test acceptance criteria to be used during the flow testing of the Nuclear Service Water Supply piping to the Auxiliary Feedwater System Pumps. This information was developed per calculation CNC-1223.42-00-0001, Rev 13A. A new test acceptance criteria sheet was originated to provide acceptable flow test pressure differentials for various flow rates through the six inch Unit 1 Nuclear Service Water System Train 1A to Auxiliary Feedwater piping from the 24 inch header to just upstream of valve 2RN250A. Meeting this criteria will ensure that Nuclear Service Water System flow to the Auxiliary Feedwater System pumps is sufficient to meet the Auxiliary Feedwater System design basis. An approved station procedure (PT/2/A/4400/014) will be used to measure and record the pressure differentials.
The design requirements of this portion of the Nuclear Service Water System are Duke Class C (Nuclear Safety Related), carbon steel, 150 psig and 150 degrees F. The new isolation valve (2RNE95) is an eight inch wafer butterfly valve. This valve is a stainless steel nuclear safety related valve with design conditions of 150 psig at 200 degrees F. and will meet the design temperature and pressure requirements of the Nuclear Service Water System as stated above. Stainless steel is an acceptable material for use in this portion of the Nuclear Service Water System. The existing vent valve is not being modified but only relocated. This valve will continue to meet the design temperature of the Nuclear Service Water System as stated above. This valve is a carbon steel nuclear safety related Class A valve with design conditions of 2735 psig at 680 degrees F. and will meet the design temperature and pressure requirements of the Nuclear Service Water System as stated above. All the new piping and fittings are nuclear safety related carbon steel and will also meet the design requirements. The new butterfly valve is suitable for use in the Nuclear Service Water System as an isolation valve. The new instrumentation configuration will

include a set of orifice flanges, an orifice plate to be installed during testing, a blank orifice plate to be installed during normal system operation, and the associated instrumentation tubing, root valves and manifold valve. Except for the two orifice plates, all these instrumentation components are nuclear safety related material and are acceptable for use in this portion of the Nuclear Service Water System. The two orifce plates are of material that is not nuclear safety related because they are not pressure boundary components, however they are acceptable for use in the Nuclear Service Water System. The addition of the weight of the new piping and instrumentation components has been evaluated for impact to the stress analysis, and support/restraint modifications are not required. Adding these valves, piping components and instrumentation components will not affect the operation or function of the Nuclear Service Water or Auxiliary Feedwater Systems during any phase of normal or accident mitigation operation. The Nuclear Service Water and Auxiliary Feedwater Systems will continue to function as described in the UFSAR and the design basis specifications.

During normal and emergency operation, the blank flow orifice is installed in the piping. The orifice plate used for testing will only be installed when this portion of the Nuclear Service Water System is out of service for flow testing which will be controlled by an approved station periodic testing (PT) procedure. Auxiliary Feedwater System flow calculations have been revised to evaluate the effect that this modification will have on the Nuclear Service Water flow to the Auxiliary Feedwater System during normal and emergency operation. Based on a calculation to compare the new resistance factor with the existing resistance factor, the impact to the flow rates was found to be negligible.

Modification CE-61479 does not involve an Unreviewed Safety Question. No changes to the Technical Specifications are required. No UFSAR changes are required.

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

Title: Minor Modification CE-61480, Provide flow measuring instrumentation, an isolation valve and a vent valve in the Nuclear Service Water System Train 2B Supply Piping to the Auxiliary Feedwater System and replace a portion of piping.

Description: Minor Modification CE-61480 and Variation Notice VN-61480E will provide flow measuring instrumentation, an isolation valve, and a vent valve in the Nuclear Service Water System Train 2B Supply Piping to the Auxiliary Feedwater System Pumps - Motor Driven 2B and Turbine Driven No. 2. Also, the modification will replace a portion of the six inch Nuclear Service Water System piping with eight inch piping. Providing this instrumentation will allow testing to be performed to determine the piping roughness of the Nuclear Service Water System piping. Replacing the six inch piping with eight inch piping results in less resistance to flow. VN-61480E provides test acceptance criteria to be used during flow testing to ensure the Nuclear Service Water System supply piping to the Auxiliary Feedwater System pumps provide acceptable flow to meet the Auxiliary Feedwater System design basis.

Evaluation: This modification affects the safety-related assured source Nuclear Service Water System Train 2B supply piping to Auxiliary Feedwater System Pumps - Motor Driven 2B and Turbine Driven No. 2. This piping will be modified to include an isolation valve (2RNE93) and flow instrumentation (2RNFE9290 and 2RNFX9290). Also, a portion of this six inch piping will be replaced with eight inch piping. There are no electrical cables or power requirements associated with this modification.

Variation Notice VN-61480E provides test acceptance criteria to be used during the flow testing of the Nuclear Service Water Supply piping to the Auxiliary Feedwater System Pumps. This information was developed per calculation CNC-1223.42-00-0001, Rev 13A. A new test acceptance criteria sheet was originated to provide acceptable flow test pressure differentials for various flow rates through the six inch Unit 1 Nuclear Service Water System Train 2B to Auxiliary Feedwater piping from the 20 inch header to just upstream of valve 2RN310B. Meeting this criteria will ensure that Nuclear Service Water System flow to the Auxiliary Feedwater System pumps is sufficient to meet the Auxiliary Feedwater System design basis. An approved station procedure (PT/2/A/4400/014) will be used to measure and record the pressure differentials.

The design requirements of this portion of the Nuclear Service Water System are Duke Class C (Nuclear Safety Related), carbon steel, 150 psig and 150 degrees F. The new isolation valve (2RNE93) is an eight inch wafer butterfly valve. This valve is a stainless steel nuclear safety related valve with design conditions of 150 psig at 200 degrees F. and will meet the design temperature and pressure requirements of the Nuclear Service Water System as stated above. Stainless steel is an acceptable material for use in this portion of the Nuclear Service Water System. The new vent valve is a one inch Y-Type globe valve. This valve is a carbon steel valve with design conditions of 2735 psig at 680 degrees F. and will meet the design temperature of the Nuclear Service Water System as stated above. All the new piping and fittings are nuclear safety related carbon steel and will also meet the design requirements. The new globe valve is suitable for use in the Nuclear Service Water System as a vent valve. The new butterfly valve is suitable for use in the Nuclear Service Water System as an isolation valve. The new instrumentation configuration will include a set of orifice flanges, an orifice plate to be installed during testing, a blank orifice plate to be installed during normal system operation, and the associated instrumentation tubing, root valves and manifold valve. Except for the two orifice plates, all these instrumentation components are nuclear safety related material and are acceptable for use in this portion of the Nuclear Service Water System. The two orifice plates are of material that is not nuclear safety related because they are not pressure boundary components, however they are acceptable for use in this portion of the Nuclear Service Water System. The addition of the weight of the new piping and instrumentation components has been evaluated for impact to the stress analysis, and support/restraint modifications are not required. Adding these valves, piping components and instrumentation components will not affect the operation or function of the Nuclear Service Water or Auxiliary Feedwater Systems during any phase of normal or accident mitigation operation. The Nuclear Service Water and Auxiliary Feedwater Systems will continue to function as described in the UFSAR and the design basis specifications.

During normal and emergency operation, the blank flow orifice is installed in the piping. The orifice plate used for testing will only be installed when this portion of the Nuclear Service Water System is out of service for flow testing which will be controlled by an approved station periodic testing (PT) procedure. Auxiliary Feedwater System flow calculations have been revised to evaluate the effect that this modification will have on the Nuclear Service Water flow to the Auxiliary Feedwater System during normal and emergency operation. Based on a calculation to compare the new resistance factor with the existing resistance factor, the impact to the flow rates was found to be negligible.

Modification CE-61480 does not involve an Unreviewed Safety Question. No changes to the Technical Specifications are required. No UFSAR changes are required.

184 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-61484, Replace, reroute, or abandon in place the existing two inch Makeup Demineralized Water System

Description: Minor Modification CE-61484 will replace, reroute, or abandon in place the existing two inch Makeup Demineralized Water System aluminium piping located in the yard between the Unit 1 Containment Mechanical Equipment Building and the Fuel Building, due to repeated failures. The aluminum piping will be replaced with polyethene piping of the same nominal size. Polyethene piping will be better suited to underground applications. This portion of the Makeup Demineralized Water System provides a source of makeup water for the Containment Chilled Water System Compression Tank located in the Containment Mechanical Equipment Building. The Makeup Demineralized Water System, the power source for the piping heat trace, and the concrete that will be removed and repaired are not required for maintenance of plant safety in the event of an accident.

Evaluation: There are no unreviewed safety questions associated with this minor modification. No Technical Specification changes are required. The modification has no effect on the probability or consequences of accidents analyzed in the UFSAR. No new accident scenarios are created by this modification. A change is required for UFSAR Figure 9-45 (piping flow drawing).

192 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-61508, Removal of Certain Recommended Action Statements from Design Basis Documents

Description: Minor Modification CE-61508 will remove certain "Recommended Action Statements" for power operated valves from Design Basis Documents for the Component Cooling System, the Fuel Pool Cooling System, the Safety Injection System, the Chemical and Volume Control System, and the Containment Penetration Valve Injection System. There are no changes to procedures or to installed plant systems, structures, or components as a result of this modification. For the Component Cooling System, Recommended Action Statements for valves KC56A, KC57A, KC81B, and KC82B when used in conjunction with the design basis and ESF signals described within each section, clearly state the effect that the valve has on the Component Cooling System as well as the Residual Heat Removal System and Containment Spray System Trains supported by the Component Cooling System, under the specified conditions of valve inoperability. The Recommended Action Statement for these valves being inoperable and in the open position provides an alternate valve alignment to prevent runout of the Component Cooling Pumps while assuring Component Cooling System Train safety functions are met. The Recommended Action Statement for valves KCC37A and KCC40B provide an alternate evaluated flowpath that satisfies both pump miniflow and pump runout concerns. Simultaneous accident alignment, Component Cooling System safety related heat removal functions, and design basis single failures have been considered. For the changes associated with the Fuel Pool Cooling, Safety Injection, Chemical and Volume Control, and Containment Penetration Valve Injection Systems; the recommended action statements do not provide any compensatory measures so no futher evaluation is required. The recommended action statements clearly state the case for system or train inoperability due to valve inoperability, and the applicable LCO must be entered. Several other Design Basis Document changes were made associated with retaining or explaining descriptive information concerning important safety related system requirements and support functions. These do not alter any safety analysis assumptions, flooding analyses, or other plant design bases.

Evaluation: There are no unreviewed safety questions associated with these changes. No change was made to any plant system, structure, component or procedure. No Technical Specification change is required. No UFSAR changes are required.

57 **Type:** Minor Modification **Unit:** 1

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

This modification 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 (0EMFT5300) 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. The total sample flow instrument no longer initiates the flow alarm. 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: Radiation Monitor 0EMF41 is not nuclear safety related. Both the new flow detector and computer are not nuclear safety related. The instruments installed by this modification provide a function previously thought to exist for Radiation Monitor 0EMF41. Post mod testing for modification 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 Radiation Monitor EMF41. A loss of power that that would disable these instruments would also disable Radiation Monitor EMF41. Blown fuses in the Radiation Monitor 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 Radiation Monitor as failures of the original instruments; however, the new instruments provide improved flow annunciation by monitoring the actual flow through the detector.

There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. This modification assigns instrument mark numbers to the new flow instruments. These new instruments have been added to the affected flow diagram for Radiation Monitor 0EMF41. UFSAR Figure 9-122 will be revised.

96 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-9307 Centrifugal Charging Pump Speed Increaser Oil Sample Valves

Description: Minor Modification CE-9307 will install an oil sample valve on the side of the Centrifugal Charging Pump Speed Increaser. An existing pipe plug will be removed and a sample valve will be installed. The sample valve and the attached piping will meet or exceed the design requirements of the present oil system for the speed increaser. The modification has been evaluated for seismic conserns,

Evaluation: There is no unreviewed safety question associated with this modification. The sample valve and attached piping have been evaluated for compatibility with the design requirements of the existing oil system. Therefore the modification will not effect the probability or consequences of accidents evaluated in the UFSAR. The modification will have no effect on the operation of the Centrifugal Charging Pumps. No Technical Specification changes are required. No UFSAR changes are required.

251 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CN-10898, Revise Hydrogen Mitigation System Description to reflect minimum current values

Description: Editorial Minor Modification CNCE-10898 was written to revise the Unit 1 and Unit 2 Electrical System Descriptions for the Hydrogen Mitigation System to include the minimum current values for each of the ignitor circuits. This documents the justification of not performing an error analysis.

Evaluation: The licensing requirements, relative to the provisions for hydrogen control, prescribed in 10CFR50.44 have evolved from deliberations among the Nuclear Regulatory Commission, the Advisory Committee on Reactor Safeguards, the NRC staff, and utilities. The NRC's requirement for ice condenser containments is that a supplemental hydrogen control system be provided so that the consequences of the hydrogen release generated during the more probable degraded core accident sequences do not involve a breach of containment nor adversely affect the functioning of essential equipment.

As part of research activities, Duke Power and other utilities investigated alternative measures of hydrogen control. As a result of these studies, a hydrogen ignition system was been installed in Catawba Units 1 and 2 to provide adequate safety margins in controlling the consequences of degraded core accidents.

This modification does not affect the Hydrogen Mitigation system as installed in the plant. There is no effect on the probability or consequences of accidents analyzed in the UFSAR. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. No UFSAR changes are required

155 **Type:** Minor Modification **Unit:** 0

Title: Minor Modifications CE-61487, CE-61487

Description: A potential adverse interaction was discovered between the Auxiliary Building Ventilation System and the Annulus Ventilation System which could impact the ability of the Annulus Ventilation System to perform its design basis functions. This potential system interaction could adversely impact the reference pressure sensed by the Annulus Ventilation System annulus pressure transmitters.

The Unit 1 and Unit 2 Annulus Ventilation System annulus pressure transmitters currently reference the annulus pressure to Elevation 577' (Train A) and Elevation 560' (Train B) electrical penetration rooms. The respective pressure transmitters are also located within the electrical penetration rooms.

To resolve this problem, a modifcation was intiated to relocate the reference leg tubing for the annulus pressure transmitters, 1(2)VEPT5000, 1(2)VEPT5001, 1(2)VEPT5010, 1(2)VEPT5011, to the Units 1 and 2 Inside Main Steam Doghouses (VE = Annulus Ventilation System, PT = Pressure Transmitter). The pressure transmitters will not be relocated (only the reference legs).

Evaluation: The Annulus Ventilation System is not an accident initiator and the proposed modifications will enhance the ability of the system to perform its design basis function (compared to the current state). The instrument reference legs will be relocated to eliminate concerns with HVAC induced problems. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. No UFSAR changes are required.

125 Type: Miscellaneous Items **Unit:** 0

Title: Auxiliary Building Ventilation System Single Train Alignment

Description: The Auxiliary Building Ventilation System is currently tested with filter units on both trains operating in parallel. A concern was identified with high airflow during single train operation. Corrective action for this concern placed administrative restrictions on the Technical Specification allowable limits for maximum air flow and carbon iodine penetration levels. Later concerns were again identified with the single train alignment. The Auxiliary Building Ventilation System was declared Operable But Degraded (OBD) with a compensatory action item issued to conservatively reduce maximum allowable Technical Specification air flow rate from 33,000 cubic feet per minute (cfm) to 32,000 cfm. In addition, this OBD placed administrative limits on the allowable maximum carbon iodine penetration level. The administrative upper limit was conservatively reduced from 4% to 3%. The proposed changes identified will allow this OBD to be deleted and will allow Auxiliary Building Ventilation System single train air flow and iodine levels to be operated with existing Technical Specification maximum allowable air flow rate of 33,000 cfm and a maximum allowable carbon iodine penetration level of 4%. Editorial changes will also be performed to the carbon sampling procedure to align notes about nominal carbon sample run-time limits with the Ventilation Filter Test Program. Other changes include a section that was added to the Auxiliary Building Ventilation System Design Basis Document to address test alignments in normal mode of operation and associated equipment required for operability.

To ensure that Auxiliary Building Ventilation System carbon filter units are tested in their most challenging configuration, it is proposed that the Auxiliary Building Ventilation System filter units be tested in a single train alignment. These tests will ensure that the Auxiliary Building Ventilation System filter units operate within the acceptable limits identified by the Technical Specification and the Ventilation Filter Test Program.

Revisions to the following documents are required:

PT/0/A/4450/001C, Auxiliary Building Filtered Exhaust Filter Performance Test, Rev 14
PT/0/A/4450/017, Safety Related Filter System Run Time Monitoring and Carbon Sampling, Rev 23
MP/0/A/7450/080, Troubleshooting and Corrective Maintenance of HVAC Dampers, Rev 4
UFSAR Section 9.4.3.2.3, Rev 10/98
CNS-1577.VA-01-0001, System VA Design Basis Document, Rev 9

Evaluation: Procedure PT/0/A/4450/001C will be revised to test Auxiliary Building Ventilation System filter trains in a single train alignment. The filter trains are currently being tested in a dual train alignment. Testing with Auxiliary Building Ventilation System filtered exhaust in a single train alignment will ensure that the Auxiliary Building Ventilation System filter units are tested in their most challenging configuration. The air flow during single train alignment will be higher, on a per train basis, than that achieved during dual train alignment. The filter unit efficiency will be tested in an alignment that provides the highest challenge to components within the filter units such as the HEPA filters and carbon adsorber. Other HVAC ventilation filtered exhaust systems (Control Room Ventilation, Fuel Pool Ventilation and Annulus Ventilation) also test the filters in a single train alignment. The proposed Auxiliary Building Ventilation System single train alignment for testing continues to meet with the testing guidelines provided by ANSI N510-80. The same test methods and instruments that are currently used will be used in the new flow configuration with the exception that B Train flow measurements will be taken at a pitot traverse downstream of the current flow monitor. This is similar to the way that A-Train flow measurements are currently taken.

This modification will reduce total Auxiliary Building Ventilation System filtered exhaust with Auxiliary Building Ventilation System Trains A and B operating in parallel. The associated negative pressure in the Auxiliary Building will be reduced. The subsequent effect on Control Room pressure will be reduced. The Control Room Ventilation System will be able to maintain the positive pressure in the Control Room more easily. Therefore this modification does not degrade the ability of the Control Room Ventlation System to maintain positive pressure in the Control Room. The Control Room pressure will be tested during this modification to ensure it remains within the allowable limits of the Improved Technical Specifications.

While testing in a single train alignment, the air flow through each filter unit will remain uniform. During LOCA operation the exhaust air flow is aligned specifically for single train operation. The long lengths of safety related filtered exhaust Auxiliary Building Ventilation System ductwork allow ample time for uniform air distribution across the ducting. Additionally the filter unit is tested with air flows of 30,000 cfm +/- 10%, while the LOCA safety alignment will exhaust only slightly more than 6,000 cfm through the Auxiliary Building Ventilation System filter units.

There are no unreviewed safety questions associated with the changes described for procedures PT/0/A/4450/001C Revision 14, PT/0/A/4450/017 Revision 23 and MP/0/A/7450/080 Revision 4. No Technical Specification changes are required. UFSAR Section 9.4.3.2.3 will be revised.

78 Type: Miscellaneous Items **Unit:** 0

Title: Change to Bases of Surveillance Requirement 3.7.5.3

Description: The Basis of Surveillance Requirement 3.7.5.3 was changed to include a statement that valves intended to supply water from the Nuclear Service Water System [1(2)RN-250A and 1(2)RN-310B] are included in the surveillance requirement.

Evaluation: This change is editorial in nature and will not affect any systems, structures, or components. The change clarifies the basis of Surveillance Requirement 3.7.5.3 to include all valves in the Auxiliary Feedwater Flowpath that do not have an auxiliary feedwater system designator. Currently these valves are tested per procedure PT/1(2)/A/4250/03D. No changes to this procedure will be required. There are no unreviewed safety questions associated with this change. No Technical Specification changes are required. No UFSAR changes are required.

152 **Type:** Miscellaneous Items **Unit:** 0

Title: Change to the Catawba Nuclear Station Pump and Valve Inservice Testing Program (Rev 24)

Description: The Catawba Nuclear Station Pump and Valve Inservice Testing Program (Rev 24), was changed to allow valves 1VG25, 1VG26, 1VG27, 1VG28, 1VG69, 1VG70, 1VG71, 2VG72, 2VG25, 2VG26, 2VG27, 2VG28, 2VG69, 2VG70, 2VG71, and 2VG72 to be tested at Cold Shutdown rather than at refueling. These valves are located in the Diesel Generator starting Air System. The valves open to admit air to the starting air distributor and close upon a sucessful diesel start. These valve were listed as cold shutdown valves and a previous 10CFR50.59 evaluation deferred the testing to refueling. Upon further review, cold shutdown testing is appropriate.

Evaluation: There are no unreviewed safety questions associated with this change. These are minor IST Program changes that do not create any new failure modes or operating characteristics. All required testing is being performed by approved procedures. No Technical Specification changes are required. No UFSAR changes are required.

132 **Type:** Miscellaneous Items **Unit:** 0

Title: Change to the Catawba Nuclear Station Pump and Valve Inservice Testing Program (Rev 24)

Description: The Catawba Nuclear Station Inservice Testing Program is being revised to incorporate the following change: Valves 1NV206, 2NV206, 1NV218 and 2NV218 will be changed from "active" to "passive". Valves NV206 and NV218 are Seal Water Heat Exchanger Inlet and Outlet Valves, respectively. The valves are a part of the Chemical and Volume Control System. Both valves are air operated plug valves. These valves are normally open to allow flow through the tube side of the heat exchanger. The valves also provide an assurred minimum flow path for the centrifugal charging pumps under normal and accident conditions. To prevent spurious repositioning of the valves they are equipped with safety related 4-way manual air valves rather than a solenoid. Since these valve are in their fail-safe position, repositioning of the valves for accident mitigation is not necessary. Since these valve are not required to move to a safe position they can be categorized as "Passive" rather than "Active".

Evaluation: This change has no effect on the overall performance of the systems involved . The change will not cause any systems to be operated outside their design limits. The change will not affect the probability or consequences of accidents evaluated in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required.

133 Type: Miscellaneous Items **Unit:** 0

Title: Change to the Catawba Nuclear Station Pump and Valve Inservice Testing Program (Rev 24)

Description: The Catawba Nuclear Station Inservice Testing Program is being revised to incorporate the following change:
For valves 1CA37, 2CA37, 1CA41, 2CA41, 1CA45, 2CA45, 1CA49, 2CA49, 1CA53, 2CA53, 1CA57, 2CA57, 1CA61, 2CA61, 1CA65, 2CA65 retain COTM (Continuous Operator Aid Computer (OAC) Temperature Monitoring) as the method for checking closure.
The valves with the "CA" designation are a part of the Auxiliary Feedwater System.
For Valves 1VG15, 2VG15, 1VG16, 2VG16, 1VG29, 2VG29,1VG30, 2VG30, 1VG31, 2VG31, 1VG32, 2VG32, 1VG59, 2VG59, 1VG60, 2VG60, 1VG73, 2VG73, 1VG74, 2VG74, 1VG75, 2VG75, 1VG76, 2VG76 defer testing from Cold Shutdown to Refueling. The valves with the "VG" designation are a part of the Diesel Generator Starting Air System. Valves 1(2)CA37, 41, 45, 49, 53, 57, 61, 65 are located on the discharge lines of the Auxiliary Feedwater Pumps. These valves open to pass Auxiliary Feedwater flow to the steam generators, if required. During normal operation, these valves close to separate high pressure and high temperature feedwater bypass or tempering flow from low pressure portions of the system. The open function is verified at cold shutdown during the Auxiliary Feedwater System flow balance. The closed function is being verified by continuous OAC temperature monitoring (COTM) for back leakage. A previous 10CFR50.59 evaluation addressed sample disassembly as the method to verify closure. Upon further evaluation, it was determined the number of sample disassembly groups was not cost effective. Poor performing valves have been changed which has led to different makes, models or styles of valve for this application. Different makes and models would necessitate a larger number of sample disassembly groups, which would require more valves per outage to be disassembled. The goal is to have all these check valves changed to be the same, which would facilitate more cost effective sample disassembly groups. COTM is adequate to verify closure, however sample disassembly supplemented with COTM is preferred.

COTM will be utilized to verify closure. For this release of the Inservice Testing Manual, Justification for Deferral (JFD) CN-CA-01 need not be revised. By utilizing COTM, only the open function needs to be deferred which is reflected in the current revision of the JFD. This evaluation returns the manual to its original state (for these Auxiliary Feedwater System valves) prior to the 10CFR50.59 evaluation which evaluated utilizing sample disassembly.

Valves 1(2) VG15,16,29,30,31,32,59,60,73,74,75,76 are located in the starting air system for the emergency diesel generator. These valves open to admit air to the starting air distributor and close to prevent fuel oil and combustion products from entering the starting air system when the diesel is running. These valves are currently listed as cold shutdown valves. The cold shutdown testing consists of isolating each bank of starting air to verify the diesel starts in the required time. Degradation in these valves would be detected by an increase in the diesel start time. Because of the complex nature of this testing, the testing should be deferred to refueling.

Evaluation: There are no unreviewed safety questions associated with these revisions to the Inservice

Testing Program. The revisions do not change the facility as described in the UFSAR, nor are there changes to the test procedures, methods, or acceptance criteria of testing already being performed for these valves. Therefore the changes are not tests or experiments, nor are they significant enough to justify inclusion in the UFSAR. No change to plant Technical Specifications is required. No UFSAR changes are required.

135 Type: Miscellaneous Items **Unit:** 0

Title: Compensatory Action for PIP 0-C98-1726 (Auxiliary Feedwater System Operability) Revision 2

Description: An operability evaluation was prepared to address an unanalyzed condition that occurred when the Upper Surge Tank (UST) temperature exceeded the Auxiliary Feedwater System suction source operability limit of 138 degrees. The high UST temperature on Unit 1 was caused by valve 1CM127, "Condensate Makeup System to Condensate Feedwater System Cleanup Flow Control Valve", opening to control flow at an incorrect flow setpoint of 14500 gallons per minute during a power decrease. The potential was identified for a single failure of this valve to quickly increase the Upper Surge Tank temperature and place the Auxiliary Feedwater system in an unanalyzed condition. A Compensatory Action was approved to isolate valve CM127 in Modes 1, 2, 3 and Mode 4 when the Steam Generators are relied upon for heat removal to prevent this unanticipated heatup of the Upper Surge Tank. This compensatory action assures Auxiliary Feedwater System operability under the conditions defined by the UFSAR, May 1997 Revision. The existing compensatory action will be revised to allow valves 1(2)CM127 to be unisolated in Modes 3 and 4 when condensate temperature would not exceed the Auxiliary Feedwater System temperature limits provided heating steam is isolated from the C feedwater heaters.

The existing compensatory action to maintain the Auxiliary Feedwater System Operable But Degraded has Operations maintain valves 1(2)CM127, "Condensate Feedwater Recirculation Control Valve", isolated in Modes 1, 2, 3 and Mode 4 when the Steam Generators are relied upon for heat removal. The purpose of this Compensatory Action is to ensure that any failure of valves 1(2)CM127 or its associated control loop which results in causing 1(2)CM127 to spuriously open does not input high flow and high energy water to the Upper Surge Tank. It is proposed to allow 1(2)CM 127 to be unisolated in Mode 3 and Mode 4 when the Steam Generators are relied upon for heat removal, provided measures are taken to ensure that no overheating of the Upper Surge Tank takes place. These measures include: isolating condensate heating sources and verifying condensate temperatures are acceptable prior to unisolating valves 1(2)CM 127. The proposed compensatory action will read as follows:

Modes 1 and 2: Maintain 1(2)CM 127 isolated by maintaining 1(2)CM126 and 1(2)CM125 closed. The C Heater Drain Pumps are not to be in service below 70% power.

Mode 3 and Mode 4 while Steam Generators are relied upon for heat removal: Maintain heating steam to C Heaters isolated. Prior to unisolating or placing 1(2)CM 127 in operation ensure C heater outlet temperature is less than 120 degrees F (OAC point C1(2)A0164 and C1(2)A0167). Ensure UST temperature remains less than 120 degrees F while 1(2)CM127 is recirculating, to the UST and isolate 1(2)CM127 if the UST temperature exceeds 120 degrees F (OAC point C1(2)A0510 and C1(2)A0511).

Evaluation: There are no Unreviewed Safety Questions associated with this revision to the original Compensatory Action. The change to the existing compensatory action is to allow valve CM-127 to be unisolated in Modes 3 and 4 provided that additional measures are taken to ensure that Upper Surge Tank water cannot be overheated. During Modes 3 and 4 with the turbine offline, the only way to heat the condensate is to supply Auxiliary Steam to the

C Feedwater Heaters. This compensatory action requires isolation of the Auxiliary Steam supply to the C Feedwater Heaters prior to unisolating valve 1CM127. The compensatory action also requires verification that condensate temperature is below 120 degrees F prior to unisolating valve CM127. This ensures that the temperature of the Upper Surge Tank will remain below 120 degrees F in accordance with the original compensatory action. This compensatory action has no effect on the probability or consequences of accidents evaluated in the UFSAR. No Technical Specification changes or UFSAR changes are required.

72 **Type:** Miscellaneous Items **Unit:** 0

Title: Compensatory Action for the Control Room Area Chillers dated 1-17-99.

Description: The Control Room Area Chiller Performance Test (PT/0/A/4450/008E) currently has three sets of acceptance criteria. The first set is for a "Fully Operable" chiller condenser and the other two sets are for "Operable but Degraded" chiller condensers. The reason for having three sets of criteria is because at different times of the year the chillers foul at different rates. Historically the condensers foul at a much greater rate during the winter months. At these times it is necessary to take credit for cooler Nuclear Service Water System temperatures in order to maintain a reasonable cleaning frequency.
A test was performed on January 17, 1999 for the B Train Control Room Area Chiller and the results of this test indicated that the chiller would have to be declared "Operable but Degraded". Conditions of operability are that the Standby Nuclear Service Water Pond temperature is less than 70 degrees F and that the Nuclear Service Water System Essential Header temperature is less than 85 degrees F in accordance with Case 2 of Engineering Calculation CNC-1211.00-00-0113, Revision 0 "Control Room Area Chiller Operability Evaluation Calculations". Therefore this compensatory action will require that the Standby Nuclear Service Water Pond temperature and the Nuclear Service Water System Essential Header temperature be monitored to ensure operability of the B Train Chiller.

Evaluation: There are no unreviewed safety questions associated with this "Operable but Degraded" evaluation. This compensatory action ensures that the Control Room Ventilation and Chilled Water Systems can perform its safety function during times when Chiller Condenser fouling rates are seasonally high. No Technical Specification changes are required. No UFSAR changes are required.

73 **Type:** Miscellaneous Items **Unit:** 0

Title: Compensatory Action for the Control Room Area Chillers dated 4-12-99.

Description: The Control Room Area Chiller Performance Test (PT/0/A/4450/008E) currently has three sets of acceptance criteria. The first set is for a "Fully Operable" chiller condenser and the other two sets are for "Operable but Degraded" chiller condensers. The reason for having three sets of criteria is because at different times of the year the chillers foul at different rates. Historically the condensers foul at a much greater rate during the winter months. At these times it is necessary to take credit for cooler Nuclear Service Water System temperatures in order to maintain a reasonable cleaning frequency. The justification for these three sets of acceptance criteria is contained in Engineering Calculation CNC-1211.00-00-0113. This calculation was recently revised to add a fourth set of acceptance criteria.

A test was performed on January 17, 1999 for the B Train Control Room Area Chiller and the results of this test indicated that the chiller would have to be declared "Operable but Degraded". Conditions of operability are that the Standby Nuclear Service Water Pond temperature is less than 77.5 degrees F and that the Nuclear Service Water System Essential Header temperature is less than 92.5 degrees F in accordance with Case 4 of Engineering Calculation CNC-1211.00-00-0113, Revision 1 "Control Room Area Chiller Operability Evaluation Calculations". Therefore this compensatory action will require that the Standby Nuclear Service Water Pond temperature and the Nuclear Service Water System Essential Header temperature be monitored to ensure operability of the B Train Chiller.

Evaluation: There are no unreviewed safety questions associated with this Operable but Degraded" evaluation. This compensatoty action ensures that the Control Room Ventilation and Chilled Water Systems can perform their safety functions during times when Chiller Condenser fouling rates are seasonally high. No Technical Specification changes are required. No UFSAR changes are required.

112 Type: Miscellaneous Items **Unit:** 2

Title: Compensatory Action for the Operable but Degraded Evaluation associated with PIP 0-C99-1675

Description: The Nuclear Service Water System to Auxiliary Feedwater System suction line testing performed on 5/4/99 determined that the Nuclear Service Water System assured source to the Auxiliary Feedwater System will not provide sufficient flow to the Auxiliary Feedwater System pump suctions under certain Chapter 15 Design Basis accident conditions. During high flow conditions of the Condensate Feedwater System or Main Steam System line break accidents, air may be admitted to the Auxiliary Feedwater System suctions due to excessive pressure drop on the Nuclear Service Water System to Auxiliary Feedwater System supply. For this reason, all three Auxiliary Feedwater System pumps on Unit 2 were declared inoperable. A compensatory action was developed to restore two motor driven pumps to operable status. The compensatory action maintains valves 2CA116A and 2CA85B closed. These are the A Train and B Train Nuclear Service Water System supply isolations to the Auxiliary Feedwater System Pump Turbine (CAPT). The CAPT is rendered inoperable by the compensatory action since the it is isolated from its assured suction source. The Operable but Degraded evaluation for PIP 0-C99-1675 has determined that the motor driven Auxiliary Feedwater System pumps will have sufficient supply pressure from the Nuclear Service Water System if the CAPT supply from the Nuclear Service Water System is isolated. The compensatory action will render the CAPT inoperable to allow the Motor Driven Auxiliary Feedwater System pumps to be restored to operable status. This removes Auxiliary Feedwater System from Technical Specification 3.7.5 action D for three pumps inoperable to action B for one pump inoperable. The compensatory action will allow the unit to be shutdown to a mode where Auxiliary Feedwater System is no longer required per Technical Specification 3.7.5 or maintain current plant mode for 72 hours until repairs can be accomplished to render Auxiliary Feedwater System fully operable in that time frame. Action D of Technical Specification 3.7.5 requires maintaining stable plant conditions until at least one Auxiliary Feedwater System pump is operable.

Evaluation: The Auxiliary Feedwater System is a nuclear safety related system which functions to remove heat from the Steam Generators and to allow cooldown of the reactor coolant system to the point where Residual Heat Removal System can be placed in service. The system is required to be operable in Modes 1, 2, 3 and Mode 4 while steam generators are utilized for decay heat removal, per Technical Specification 3.7.5. This function may be required for normal operation and for UFSAR Chapter 15 Accident Mitigation. The Auxiliary Feedwater System by design has five suction sources which may be aligned. These are:

1. Auxiliary Feedwater Condensate Storage Tank (CACST)	Non-Safety Related
2. Upper Surge Tank (UST)	Non-Safety Related
3. Condenser Hotwell	Non-Safety Related
4. Nuclear Service Water System	Safety Related
5. Condenser Cooling Water System	Non-Safety Related

The Auxiliary Feedwater System is designed so that the normal sources are the non-safety condensate grade sources from the CACST or the UST. Currently, the Auxiliary Feedwater Storage Tank (CACST) is normally isolated. Therefore, the Upper Surge Tank is currently the normal source for the Auxiliary Feedwater System. Under accident conditions if the non-safety sources are not available, the suction will automatically swap to the assured source, the Nuclear Service Water System. Non-safety suction source availability is determined by having adequate pressure from these sources on safety related pressure switches. When a two out of three low suction pressure is detected with an autostart signal present the Auxiliary Feedwater System suction will automatically swap to the Nuclear Service Water System assured source. Valves CA116A and CA85B automatically open on Auxiliary Feedwater System low suction pressure with a Auxiliary Feedwater System auto start signal present to supply the CAPT from its assured source. If Auxiliary Feedwater System has been reset, the Auxiliary Feedwater System pumps will automatically trip on low suction pressure.

This compensatory action will maintain valves 2CA116A and 2CA85B closed. The compensatory action renders the CAPT inoperable since the assured Nuclear Service Water System source is isolated. The Motor Driven Pumps are restored to operable status per the Operable but Degraded evaluation because the maximum flow demand during the limiting main steam line break accident will be reduced by the isolation of the CAPT from the Nuclear Service Water System. The Auxiliary Feedwater System accident analysis assumes the worst case Auxiliary Feedwater System failure. The limiting Auxiliary Feedwater System failure for the feedwater line break and loss of feedwater accidents where Auxiliary Feedwater System is credited is the CAPT. Therefore the motor Driven Auxiliary Feedwater System pumps are fully capable of performing the design function of the Auxiliary Feedwater System system. With the CAPT inoperable, the plant will be operated within the Limiting Conditions for Operation (LCO) specified by the Technical Specifications. Rendering the CAPT inoperable in order to restore the Motor Driven Auxiliary Feedwater System pumps is bounded by the accident analysis.

The CAPT will still perform its beyond Chapter 15 design functions to mitigate loss of all AC power since the condenser circulating water supply to the CAPT is not affected by the compensatory action.

The Auxiliary Feedwater System does not initiate any accidents evaluated in the SAR. In the current plant configuration, with Nuclear Service Water System supplying all three Auxiliary Feedwater System pumps, the three Auxiliary Feedwater System pumps cannot supply the S/G's in the event of a Main Steam System or Condensate Feedwater System line break with a failure of a Nuclear Service Water System train and depletion of the of the non safety condensate grade sources. The Nuclear Service Water System piping has degraded to the point it cannot supply all three Auxiliary Feedwater System pumps in events resulting in high Auxiliary Feedwater System flowrates. This compensatory action assures that the Auxiliary Feedwater System can meet its design function in all accidents under the assumptions for the LCO for one Auxiliary Feedwater System pump inoperable. This compensatory action does not increase the probability of an accident evaluated in the SAR. The compensatory action assures the Auxiliary Feedwater System can perform its design function to mitigate accidents within the constraints of the Technical Specification LCO for one inoperable Auxiliary Feedwater System pump.

Isolation of the Nuclear Service Water System source to the CAPT increases the probability of damage to the CAPT slightly in the event of a non-safety condensate source piping failure or depletion. However, the CAPT is declared inoperable with the compensatory action in place and the plant will be operated under the assumptions and constraints of Technical Specification 3.7.5 LCO B. In the event the Condensate Storage and Condenser Cooling Water sources fail or are depleted, the CAPT would most likely seize and could possibly generate missiles. Per the Auxiliary Feedwater System Design Basis Document, the Auxiliary Feedwater System pumps are located in separate pits for protection from CAPT generated missiles. The CAPT will be declared inoperable, therefore it is not required to perform a safety function within the restrictions of the LCO. The probability of malfunction of the motor driven pumps is reduced by this compensatory action that assures that the operable pumps will not air bind due to a loss of normal Auxiliary Feedwater System suction sources.

It is concluded that there are no unreviewed safety questions associated with this Operable but Degraded Compensatory action. No Technical Specification changes are required. No UFSAR changes are required.

113 **Type:** Miscellaneous Items **Unit:** 1

Title: Compensatory Action for Unit 1 leakage testing of valve 1RF-389B

Description: Isolation of valve 1RF-389B for Containment Penetration Valve Injection System leakage testing will isolate the following items that are committed fire protection items per the Selected Licensee Commitment Manual: Fourteen fire hose racks in the Auxiliary Building, nine fire hose racks in the Unit 1 Containment , and three sprinkler systems (Unit 2 Battery Room Corridor, Unit 2 Cable Room Corrider, and the Unit 2 Auxiliary Feedwater Pump Room). This compensatory measure is intended for use with the Unit in No-Mode and is only applicable to the fourteen hose racks in the Auxiliary Building. The isolation of the nine hose racks in Containment only requires a compensatory measure during Modes 5 and 6 and does not require a compensatory measure during no-Mode. Compensatory measures in accordance with SLC 16.9-2 will be established for the three sprinkler systems isolated by the test alignment.

Evaluation: There is no unreviewed safety question associated with this activity. The probability or consequences of an Auxiliary Building fire will not be increased . The compensatory action will ensure that adequate manual fire fighting capabilities can be restored in a timely manner to support fire brigade response. No Technical Specification changes are required. No UFSAR changes are required.

65 Type: Miscellaneous Items **Unit:** 1

Title: Compensatory Action for Unit 1 Reactor Building Hose Rack Header Isolation/Impairment

Description: Catawba Site Directives require backup fire suppression equipment to be provided during Mode 5 and Mode 6 any time the Unit 1 Reactor Building hose rack header has been removed from service. Based on a review of the fire hazard identified in containment, the backup fire suppression requirements are determined to be as follows:

1. Provide an additional 450 feet of fire hose at Fire Hose Station 1RF489. This fire hose will provide backup fire suppression capability for fighting fires within the Unit 1 Lower Containment.
2. Provide an additional 300 feet of fire hose at Fire Hose Station 1RF265. This fire hose will provide general fire suppression capability within the Unit 1 Upper Containment.

The additional fire hose will be dedicated for fighting fires inside the Unit 1 Upper and/or Lower Containment. The Fire Brigade Incident Commander has the option to request additional fire hose if needed.

Evaluation: The actions required by this compensatory measure will ensure adequate response to a containment fire and the availability of adequate fire fighting capabilities for the identified fire hazards. This compensatory measure will not increase the probability of accidents or increase the consequences of accidents 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.

66 **Type:** Miscellaneous Items **Unit:** 2

Title: Compensatory Action for Unit 2 Reactor Building Hose Rack Header Isolation/Impairment

Description: Catawba Site Directives require backup fire suppression equipment to be provided during Mode 5 and Mode 6 any time the Unit 2 Reactor Building hose rack header has been removed from service. Based on a review of the fire hazard identified in containment, the backup fire suppression requirements are determined to be as follows:

1. Provide an additional 450 feet of fire hose at Fire Hose Station 1RF485. This fire hose will provide backup fire suppression capability for fighting fires within the Unit 2 Lower Containment.
2. Provide an additional 300 feet of fire hose at Fire Hose Station 1RF233. This fire hose will provide general fire suppression capability within the Unit 2 Upper Containment.

The additional fire hose will be dedicated for fighting fires inside the Unit 2 Upper and/or Lower Containment. The Fire Brigade Incident Commander has the option to request additional fire hose if needed.

Evaluation: The actions required by this compensatory measure will ensure adequate response to a containment fire and the availability of adequate fire fighting capabilities for the identified fire hazards. This compensatory measure will not increase the probability of accidents or increase the consequences of accidents 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.

139 **Type:** Miscellaneous Items **Unit:** 0

Title: Compensatory Action Guidelines - Plant Access Doors Revision 16

Description: Revision 16 of the Compensatory Action Guidelines Plant Access Doors was initiated to incorporate a variety of changes associated with Auxiliary Building Doors. These changes included:

1. Adding doors AX314 #3, AX513B, AX533C and AX656. The addition of these doors reflects changes to address concerns with potential problems associated with control of the pressure within the annulus as identifed by Corrective Action Program Report (PIP) 1-C99-2001.
2. Revision to Doors AX352B and AX393C to also reflect changes to address concerns identified by PIP 1-C99-2001.

Revision 16 affects only the information associated with the doors listed above.
The changes in revision 16 add four doors and revise the noted associated with two doors already in the compensatory action. All of these doors need to be maintained in the open position to allow pressure to equalize between the electrical penetration rooms and the stairwells. These doors serve as security and fire doors. The reason that it is acceptable to leave the doors open from a security perspective is that a security guard will be posted at the door when it is open. Should a design basis accident occur that causes the guard to have to leave his position, the security boundary will be moved back to a safe position while still maintaining secure control over the door and electrical penetration rooms. The reason it is accepotable to leave the doors open from a fire barrier standpoint is that a fire watch will be established to protect the area from a potential fire. Should the design basis accident occur the door can be left open since a fire and design basis accident are not considered to occur concurrently. If a fire occurred it is assumed that it would be extinguished prior to any design basis accident.

Evaluation: These Compensatory Action Guidelines serve as an overall review of doors within the Auxiliary Building between the "AA" and "QQ" walls. The guidelines identify the design features of each door and allow activities which could prevent the door from closing in its normal manner to occur if the design feature is not impaired. If the design feature is impaired then compensatory actions can be put in place. These compensatory actions are limited to fire boundary doors which require a firewatch, tornado doors which must be closed within one hour and Security doors which require security access control to be established. With the requirements of the Compensatory Actions satisfied, no unreviewed safety questions would exist. No Technical Specification changes are required. No UFSAR changes are required.

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

Title: Compensatory Action Guidelines - Plant Access Doors Revision 17

Description: Revision 17 of the Compensatory Action Guidelines for Plant Access Doors incorporates two doors that have not previously been included in the compensatory action. An editorial change was made throughout the compensatory action to change references to "Station Directive 2.12.7" to "Nuclear System Directive 316". The two doors that were added were AX217F and AX260F. These doors are located in the floor (horizontal position) on the 543' elevation of the plant. These doors lead into the Unit 1 and Unit 2 Turbine Driven Auxiliary Feedwater Pump Pits. The addition of these doors does not change the intent of the original 10CFR50.73 evaluation that was done for the Compensatory Action Program for plant access doors.

Evaluation: Plant access doors may serve any combination of the following nuclear safety related functions: Fire Door, Security Door, Tornado Missle Barrier, Tornado Pressure Door, Ventilation System Boundary, Environmental Qualification Zone Barrier, Carbon Dioxide Fire Suppression System Zone Boundary. Each of these design functions is evaluated and it is concluded that no unreviewed safety questions exist when the specified compensatory action is in place. The two doors added per this revision serve to provide fire protection by creating a fire barrier and serve as a halon boundary for the Turbine Driven Auxiliary Feedwater Pump pits. The compensatory action for these doors ensures that if the doors need to be held open, fire watch is posted in the area. No Technical Specification changes are required. No UFSAR changes are required.

188 **Type:** Miscellaneous Items **Unit:** 0

Title: Compensatory Action Guidelines - Plant Access Doors Revision 18

Description: Revision 18 of the Compensatory Action Guidelines - Plant Access Doors was revised to include the following two cases:

1. The way security compensatory actions apply to doors AX352D and AX393D was changed. These doors are the lower airlock doors on Unit 1 and Unit 2 respectively. These doors were originally the security barriers for access into Lower Containment. This requirement applied no matter what operational mode the unit was in. The security requirements were changed such that during Modes 5, 6, and No Mode, other doors become the security barrier for lower containment. The other doors are AX352B and AX393C (for Unit 1 and Unit 2 respectively). The Compensatory Action was revised to show that the lower cointainment doors are not security barriers if the affected Unit is in Modes 5, 6 or No Mode.
2. Potential for confusion was identified associated with a previous revision of the compensatory action guidelines. The previous revision (Revision 16) was initiated to allow certain doors to be propped open to address a previously identified concern. The need to prop the doors open no longer exists, therefore to reduce confusion changes made per revision 16 will be deleted. Doors affected by this change are AX314 #3, AX513B, AX533C, AX656, AX352B, and AX393C.

Evaluation: Plant access doors may serve any combination of the following nuclear safety related functions: Fire Door, Security Door, Tornado Missle Barrier, Tornado Pressure Door, Ventilation System Boundary, Environmental Qualification Zone Barrier, Carbon Dioxide Fire Suppression System Zone Boundary. Each of these design functions was evaluated and it was concluded that there are no unreviewed safety questions associated with the two changes incorporated in Revision 18. No Technical Specification changes are required. No UFSAR changes are required.

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

Title: Compensatory Action Guidelines - Plant Access Doors Revision 19

Description: Revision 19 of this compensatory action was initiated to incorporate changes associated with cleaning portions of the Liquid Waste Recycle System. The Waste Evaporator Feed Tank and the Laundry and Hot Shower Tank are to be cleaned and drained per Work Order 97049829. In order to do this a hose will be routed from the tanks into the duct shaft on elevation 543 at colum MM-54/55. The door to this duct shaft is AX227D. The "Compensatory Action Guidelines - Plant Access Doors was revised to add this door.

Evaluation: Plant access doors may serve any combination of the following nuclear safety related functions: Fire Door, Security Door, Tornado Missle Barrier, Tornado Pressure Door, Ventilation System Boundary, Environmental Qualification Zone Barrier, Carbon Dioxide Fire Suppression System Zone Boundary. Each of these design functions was evaluated and it was concluded that there are no unreviewed safety questions associated with the change incorporated in Revision 19. No Technical Specification changes are required. No UFSAR changes are required.

218 **Type:** Miscellaneous Items **Unit:** 0

Title: Compensatory Action Guidelines for Firestop Penetration AX-635-W-0635 per Work Order 97049829-08

Description: Compensatory Actions were established for breaching Firestop Penetration AX-635-W-0635. This penetration is a Selected Licensee Commitment (SLC) Fire Barrier and a UFSAR committed tornado pressure boundary. The work activity involved is the cleaning of Tank 1WLTK00339. A two inch stainless steel pipe will be installed 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. The arrangement is satisfactory for a fire protection seal based on fire protection specification DPS-1435.00-00-0002. 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 3 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 the two ends of the pipe at all times. The hose is rated for 225 psi internal pressure and can easily withstand the 3 psi tornado external depressurization. There are no ventilation or security concerns since the Auxilairy Building Ventilation System serves 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 issue. There is no CO2 Fire Suppression Boundary involved.

The compensatory actions required cover only the period while the stainless steel piping is being installed through the partially removed fire penetration.

The compensatory actions consist of the following:

1) A fire watch is required to be placed in accordance with Nuclear System Directive 316 and SLC 16.9-5 anytime either of the blind flanges are removed.
2) Provisions must be made to reinstall the fire penetration and at least one pipe cap (for the sitiuation where the pipe is left in the penetration) within one hour of a tornado watch or warning in York County.

Evaluation: There are no unreviewed safety questions associated with this Compensatory Action. The reliability of the existing barrier will be maintained by this compensatory action. This compensatory action will not affect the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required.

59 Type: Miscellaneous Items **Unit:** 0

Title: Compensatory Action Instructions for Modification and/or repair of Control Room Pressure Boundary, Revision 11

Description: 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. The Control Room Pressure Boundary Compensatory Action allows for modifications and/or repairs to the Control Room pressure boundary without rendering the Control Room Ventilation System inoperable. The Compensatory action gives adequate guidance to ensure that the Control Room pressure boundary integrity is restored within five minutes of a LOCA or fire or chemical release on site. The measures outlined in the Compensatory Action ensure that the Control Room Operator dose stays within the limits given in General Design Criterion 19.

Evaluation: There is no unreviewed safety question associated with this compensatory action. The Control Room Ventilation System could still perform its design basis function. The following items were considered: Safe Shutdown Earthquake, Single Active Failure, Design Basis Tornado, Chlorine Release, Combustion Particle Release, Airborne Contamination (Dose) Analysis, Fire Door Functions of the Control Room Doors, Security Access Control Functions. No Technical Specification changes are required. No UFSAR changes are required.

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

Title: Compensatory Actions for Hydrogen Skimmer System Suction valves for current operability of the Hydrogen Skimmer System

Description: An operability evaluation for Problem Investigation Process Serial Number 0-C98-0578 determined that the Hydrogen Skimmer System inlet isolation valves were operable but degraded. Technical Specification Surveillance Requirement 3.3.1.10 implies that these valves should be fully open within 600 seconds of receipt of a Containment High-High Pressure (Sp) signal. The previous interpretation of this time requirement was that the valves only needed to begin to open within 600 seconds. The time delay relays for the valves were reset so that the valves would be fully open within the required 600 second time frame. A Compensatory Measure was implemented to place calibration procedures IP/1(2)/A/3173/05 and Inservice Test (IWV) Procedures PT/1(2)/A/4200/36 on hold until the procedure or acceptance criteria can be revised with new limits to ensure continued operability.

Evaluation: There is no unreviewed safety question associated with this compensatory measure. It provides a temporary method to ensure that Surveillance Requirement 3.3.1.10 is complied with. This compensatory measure will have no adverse effect on compliance with Technical Specification 5.5.8 because the maximum allowable IWV stroke times defined in this compensatory measure are less than the 666 second acceptance criteria for stroke time of the valves. This compensatory action will not adversely impact post accident operation of the Hydrogen Skimmer System, Containment Air Return Fan System, or any other nuclear safety related equipment required to mitigate the consequences of any design basis accident. This compensatory measure will not impact any other containment peak pressure or temperature calculations or analyses. No Technical Specification changes are required. No UFSAR changes are required.

42 Type: Miscellaneous Items **Unit:** 0

Title: Compensatory Actions required by an "Operable but Degraded" operability evaluation associated with Problem Investigation Process Report 0-C98-1726

Description: The need was identified to evaluate the potential single failure of the non-nuclear safety related valve CM127 for both Units. Valve CM127 is used during normal full power operation to provide mini-flow protection for the non-nuclear safety related Condensate Booster Pumps. During a power reduction this valve 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. The elevated temperatures resulted in declaring all three Auxiliary Feedwater Pumps inoperable. This compensatory action is a temporary measure to prevent recurrence of the event. The evaluation addresses the effect of this flow path on 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. The Compensatory Action to maintain the Auxiliary Feedwater System in an "Operable but Degraded" condition will be imposed any time the Unit is in Modes 1, 2, or 3 and has the following requirements: 1) Isolate the Condensate Makeup recirculation flowpath to the Upper Surge Tank by closing valves CM128 and CM125 in Modes 1, 2, and 3. 2) The C Heater Drain Pumps are not to be in service below 70% power. The purpose of the compensatory action is to ensure that a single failure of the non-nuclear safety related component CM127 does not input high flow and high energy water to the Upper Surge Tank. The last item of the Compensatory Action minimizes the potential for a low-flow Condensate Booster Pump trip if the reactor trips below the P-9 setpoint.

Evaluation: This compensatory action may affect two accidents described in the UFSAR. These are (1) a feedwater system pipe break and (2) a loss of feedwater. An evaluation concluded that the plant would operate within the limitations of the piping design and the ANSI B31.1 Code. Therefore the Feedwater System Pipe Break accident is not affected. The evaluation stated that the operation with the CM127 flowpath isolated may result in a loss of Condensate Booster Pumps during a loss of stator cooling runback. A Condensate Booster Pump trip under these conditions may result in a trip of the Main Feedwater Pumps. Therefore, a stator cooling runback could result in a loss of feedwater. The evaluation stated that the only loss of stator cooling runback which has been experienced at Catawba resulted in a reactor trip due to loss of both Main Feedwater Pumps. This parallels the SER acknowledgement that Stator Cooling malfunctions may result in a Turbine Trip. It is acknowledged that Overpower Delta Temperature (OPDT) and Overtemperature Delta Temperature (OTDT) runbacks are also capable of continuously running the turbine back to power levels below 40%. However, there is no credible scenario which would cause a continuous OPDT or OTDT runback for any significant power range. All of these runbacks that have occurred at Catawba have been of very short duration affecting power levels by only a few percent. This Compensatory Action will not affect the initiation of any runbacks for any reason therefore the frequency of runbacks is not affected. Therefore, the probability of the loss of feedwater accident or the feedwater system pipe break accident evaluated in the UFSAR is not increased. No

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

207 **Type:** Miscellaneous Items **Unit:** 0

Title: Condition of Operability associated with PIP 0-C97-3621

Description: The administrative limit on dose equivalent iodine (I-131) (DEI) specific activity in the Reactor Coolant System is being amended from the current restrictions. The following administrative limits are in place for the Reactor Coolant System:

1. Equilibrium DEI specific activity in the Reactor Coolant System shall not exceed 0.099 microcuries/gm (this restriction is unchanged from previous evaluations).
2. Equilibrium DEI specific activity in the Reactor Coolant System shall not exceed 0.064 microcuries/gm when the letdown flow rate is greater than 80 gallons per minute (gpm) during unit operations as outlined in Tech Spec 3.4.16 (this restriction is new).
3. Chemical and Volume Control System letdown flow rate shall not exceed 125 gpm during unit operations as outlined in Tech Spec 3.4.16 (this restriction is new).
4. Transient DEI specific activity in the Reactor Coolant System shall not exceed 15 microcuries/gm (this restriction is unchanged from previous evaluations).
5. Equilibrium DEI specific activity in the S/G secondary side shall not exceed 0.055 microcuries/gm (this restriction is unchanged from previous evaluations).

Evaluation: The limits on DEI specific activity are conditions of operability for the Reactor Coolant System. These conditions were determined in an operability evaluation for Corrective Action Program Report Serial Number 0-C97-3621. The upper limit on specific activity, along with the restrictions on maximum allowable letdown flow rate, is being revised to account for the fact that the iodine spiking production rate is defined as a function of system losses and radiological decay. Reactor Coolant System letdown flow rate is an input to this nuclide production rate during an iodine spiking event. Letdown flow in excess of 80 gpm has not been accounted for in the analysis of radiological consequences from a postulated Steam Generator Tube Rupture Accident.

In Revision 1 to Generic Letter 91-18, the NRC has set forth guidelines for the evaluations of compensatory measures for degraded and non-conforming conditions pursuant to 10 CFR 50.59. In these guidelines, the NRC has agreed that "the intent of the 10 CFR 50.59 evaluation is to determine whether the compensatory action itself (not the degraded condition) impacts other aspects of the facility described in the SAR. The NRC further stated that "in considering whether a compensatory measure may affect other aspects of the facility, a licensee should pay particular attention to ancillary aspects of the compensatory measure that may result from actions taken to directly compensate for the degraded condition." Pursuant to these guidelines, the "degraded conditions" (i.e., the potential inadequacy of the limit on equilibrium DEI specific activity in the Reactor Coolant System outlined in the plant Technical Specifications) will not be considered in the safety review and USQ evaluation of the condition of operability.

The new restrictions (numbers 2 and 3) provide additional assurance that an allowable combinations of letdown flow rate and Reactor Coolant System leakage are addressed in the dose consequences for the SGTR. It is possible to have greater letdown flow rate than addressed in this evaluation, but not allowed under condition 3 of this set of restrictions. With these limits in place, the dose consequences are below the guideline values, and hence, there is no change in the consequences of any accidents.

There are no physical changes, modifications or ancillary effects on plant systems, structures or components resulting from these restrictions. The restrictions are entirely administrative in nature, and as such, no new failure modes are created, no common failure modes are identified and no new accidents are created.

The condition of operability only sets an upper limit on DEI specific activity in the Reactor Coolant System, and does not affect the operation of any plant equipment. None of the issues addressed herein constitute accident initiators.

The new limit on equilibrium DEI specific activity in the Reactor Coolant System is less than the limit assumed in the analysis of the MSLB, Small Line Break and SGTR Accidents with the accident initiated iodine spike. Therefore, the consequences of these accidents are not increased with this activity.

No new accidents are identified, since this condition of operability does not affect any plant components or the operation of any plant SSCs.

No changes to SSCs were identified which would cause an increase in probability of malfunction of SSCs important to safety. There are no modifications to plant SSCs associated with this condition of operability.

No common failure modes are created. No new failure modes are identified.
No change was identified which causes a decrease in fission product barrier integrity or performance. The integrity of the cladding is not adversely affected by this condition of operability.

These conditions of operability do not create any unreviewed safety questions or necessitate any Technical Specifications amendments. UFSAR revisions associated with this issue will proceed per the UFSAR update process. For the radiological consequences associated with the SGTR accident, a license submittal is planned for the beginning of the year 2000. This license amendment will address the effect of Reactor Coolant System leakage and letdown flow rate on iodine spiking. The restrictions implemented under this evaluation will not become part of the UFSAR.

208 **Type:** Miscellaneous Items **Unit:** 0

Title: Condition of Operability associated with PIP 0-C98-4871

Description: A new limit on equilibrium I-131 Dose Equivalent (DEI) specific activity in the Reactor Coolant System is being set. The limit is 0.099 microcuries/gm. This limit was determined as condition for operability in Problem Investigation Process (PIP) Serial Number 0-C98-4044. The concern of that report was that a term associated with Reactor Coolant System leakage had been omitted from the calculation of the production rate used in the analyses of radiological consequences of accidents with coincident iodine spike.

Evaluation: The limit on DEI specific activity is less than (1) the corresponding values assumed in the safety analyses of all associated design basis events, (2) the limits of Technical Specification 3/4.4.8 and (3) the license conditions associated with the safety evaluation for Facility Evaluation Amendment 159/151. Neither any plant equipment nor operation of any plant equipment is affected by this condition of operability. There is no unreviewed safety question associated with this condition for operability. No Technical Specification change is required. No UFSAR change is required.

114 **Type:** Miscellaneous Items **Unit:** 0

Title: Control Room Ventilation System HEPA and/or Carbon Filter Replacement Compensatory Action, Revision 1

Description: Control Room Ventilation System HEPA and/or Carbon Filter Replacement Comp Action (November 19,1998 Revision 1). This compensatory action applies to the replacement of upstream HEPA filters or carbon media in the Outside Air Pressurizing Filter Train of the Control Room Ventilation System. After replacing either the upstream HEPA or carbon media in these filter units, there is some time period prior to the retest of the filters that the filters will be placed in service in order to perform a retest. If a design basis accident were to occur during this time period, the plant could possibly be outside of the dose analysis. Therefore, this compensatory action will be used to ensure that the plant stays within its dose analysis prior to declaring the filter unit operable by shutting down the tested filter unit in the event of an accident.

Evaluation: There is no unreviewed safety question associated with this compensatory action. Per UFSAR 6.4.1 the design bases of the habitability system for the control room includes the capability to: withstand a safe shutdown earthquake, function properly following any single active failure, function during a design basis tornado, to detect and limit concentrations of chlorine gas or products of combustion entering the control room, to shield control room operators from radiation sources, to detect and limit the introduction of airborne radioactive contamination into the control room such that exposure to personnel will not exceed the specified limits, and to permit safe shutdown of the plant from the control room following a Loss of Coolant Accident.

Of the above design bases, an inoperable filter train in operation could affect the ability to limit airborne radioactive contamination. The ability to detect these contaminants is not affected due to the intake monitors being located upstream of the Outside Air Pressurizing Filter Train.

Control Room habitability is required for all potential design basis events which release radioactivity to the environment. These events are defined in the UFSAR as follows: Main Steam Line Break, Loss of Power Rod Ejection Accident, Instrument Line Break, Steam Generator Tube Rupture, Loss of Coolant Accident, Waste Gas Decay Tank Rupture, Liquid Storage Tank Rupture, Fuel Handling Accident Outside Containment, Fuel Handling Accident Inside Containment.

The LOCA accident is the limiting design basis event for Control Room Dose for normal operation of the Control Room Ventilation System and is the only event which has been rigorously analyzed relative to control room operator dose. Per the Compensatory Action Instructions, the dose analysis for the control room operators could be affected by having an inoperable filter train in service. However, by taking credit for:
Current ILRT test results being less than those assumed in the accident analysis, Control Room Ventilation System Flow rates being balanced in such a manner that Iodine Protection Factors are greater than or equal to 80 and Pressurization flow rates are less than or equal to 1000 cfm at 1/8 inwg and ECCS leakage less than or equal to 1.0 gpm, and ensuring that the inoperable filter train is secured within five minutes of an accident, the Control Room Operator dose stays within GDC 19 limits. The affected train of the Control Room Ventilation/Chilled Water system has been declared inoperable and the

action statement for Technical Specification 3.7.6 has been entered.

The Control Room Ventilation System, along with the Control Room Area Chilled Water System are designed to maintain a suitable environment in the control room and control room areas for the operation of unit and plant controls. Specifically it is provided to maintain ambient air temperature within the continuous duty rating for equipment and instrumentation in these areas. The operation of one train of the Control Room Ventilation System is sufficient to meet these requirements. It is determined that the performance of the operable Control Room Ventilation System train is not degraded either with the test activities of the affected train or with the compensatory action.

Although the Control Room Ventilation System has been identified as possibly causing a spurious Safety Injection in the event of losing both trains of the Control Room Ventilation System due to control room heatup, this compensatory action will not affect the cooling capability of the system. The probability of a malfunction of equipment important to safety will not be increased because this compensatory action does not affect any system support functions. The consequences of an accident previously evaluated in the SAR will not be increased because the compensatory actions as identified will ensure that Control Room Operator doses during any of the design basis events will not increase above the dose analysis limit. The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased due to the compensatory measures outlined. The ability of the operable train to filter air passed to the control room and maintain the required positive pressure is not degraded with this compensatory action. Also, one train of the system has been declared inoperable. The equivalent failure of one complete train of Control Room Ventilation has already been evaluated. The possibility of an accident different from one already evaluated in the UFSAR will not be created because this compensatory action has no effect on the ability of the system to maintain temperature in the control room within the acceptable limits. The possibility of a different type of malfunction of equipment important to safety than already evaluated in the SAR will not be created since this compensatory action merely instructs Operations personnel to secure the inoperable filter train if an accident occurs. This will ensure that operator doses stay within the dose analysis assumptions.

The margin of safety to any Technical Specification Bases for the Control Room Ventilation System will not be decreased due to the compensatory measures outlined. The Technical Specification bases applicable to the control room ventilation system are temperature in the control room which is not affected by this compensatory measure and control room habitability requiring operator doses to stay below GDC 19 limits which this compensatory measure assures. This compensatory measure does not affect any of the Technical Specification basis for the Auxiliary Building Ventilation System. Additionally, this compensatory measure does not affect any of the bases for the Chlorine Detection Systems Technical Specification. No Technical Specification changes are required. No UFSAR changes are required.

142 **Type:** Miscellaneous Items **Unit:** 0

Title: Generic Compensatory Action Guidelines for Plant Hatches, Revision 6

Description: This revision to the Generic Compensatory Action Guidelines for Plant Hatches addresses the removal of the hatch above the Unit 1 B Train Centrifugal Charging Pump in Mode 1-4. The removal of this hatch during modes 1-4 is required to support the maintenace activities associated with the pump failure which occurred on 6-11-99.

Evaluation: There is no unreviewed safety question associated with revision 6 to the Generic Compensatory Action Guidelines for Plant Hatches. Hatches are not accident initiators. No Technical Specification changes are required. This activity violates the Technical Specification on Auxiliary Building Ventilation however a NOED was obtained. No UFSAR changes are required.

45 Type: Miscellaneous Items **Unit:** 0

Title: Improved Technical Specification Bases Changes

Description: Since implementation of the Improved Technical Specifications (ITS) at Catawba, a number of changes have been identified to the ITS Bases. The following ITS Bases changes are covered under this evaluation:

a. Change to ITS Bases B 3.4.9 (Pressurizer) Surveillance Requirements section to modify SR 3.4.9.2 to reflect the actual testing method that is used in the plant for verifying pressurizer heater capacity.

b. Change to ITS Bases B 3.7.5 (AFW System) LCO section to include operability aspects of the Nuclear Service Water System assured water source on the required trains of the Nuclear Service Water System. The proposed change clarifies that a single inoperable the Nuclear Service Water System source to the Auxiliary Feedwater trains only requires that one Auxiliary Feedwater train be considered inoperable.

c. Change to ITS Bases B 3.8.4 (DC Sources - Operating) Actions section to modify Actions A.1 and A.2 to allow the spare charger to be substituted for the primary charger. The proposed change clarifies that using the spare charger maintains a fully operable DC source.

d. Not used.

e. Change to ITS Bases B 3.9.3 (Containment Penetrations) Background section to delete discussion material referring to "containment closure." This discussion is editorially inconsistent with the Catawba containment closure process, which is the process used to ensure containment operability. The Bases refers to containment closure in the sense that all potential escape paths are closed or exhausting through an operable containment purge exhaust HEPA filter and charcoal adsorber. This change does not affect any technical requirements of this Bases section.

f. Change to ITS Bases B 3.6.12 (Ice Bed) Surveillance Requirements section (SR 3.6.12.2) to include clarifying wording from Catawba's old Technical Specifications (prior to ITS) that indicates that multiple discrepant flow channels in an ice condenser bay represents evidence of abnormal degradation of the ice condenser. Catawba presently interprets the existing ITS Bases wording that says that more than one discrepant flow channel in a bay is unacceptable to mean that abnormal degradation exists. Catawba does not interpret the existing wording to mean that the ice bed is inoperable. A review of the ITS conversion submittal section that pertains to this Surveillance Requirement indicates that no technical changes were made for SR 3.6.12.2. Catawba had always interpreted the old TS for this surveillance to mean that evidence of abnormal degradation of the ice condenser did not automatically render the ice condenser inoperable.

Evaluation: While these Bases changes involve Bases pertaining to a number of plant structures, systems, and components (SSCs), in no cases do the changes result in any changes to the plant or the way it is operated or maintained. No SSC is modified or caused to operate in a different manner from current operation. No seismic, environmental, materials, or reactivity effects are created on these SSCs as a result of the proposed Bases changes. No new failure modes/effects or new types of system/component interactions will be introduced upon any SSC as a result of these changes. SSC behavior during steady state and transient conditions will be unaffected by any of these Bases changes. Plant response to any external phenomena, natural or man-made, likewise will not be impacted. No plant transient 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 SSCs are being maintained. No actual regulatory commitments are being eliminated or reduced by these changes. There are no unreviewed safety questions associated with these ITS Bases changes. No Technical Specification changes are required. No UFSAR changes are required.

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

Title: Improved Technical Specification Bases Changes

Description: During the latter stages of ITS implementation at Catawba, several errors and needed editorial enhancements were noted in various ITS Bases sections. These errors and needed editorial enhancements were noted after all NRC review of the ITS submittal was complete and the NRC Safety Evaluation for the ITS amendment issued.

The following ITS Bases changes are covered under this change package:

1. Bases Section B 3.2.4, QPTR, Applicability section: The discussion regarding the note that states that the LCO is not applicable until the excore nuclear instrumentation is calibrated subsequent to a refueling was editorially revised. This revision clarifies that the calibration that is referenced here is the final calibration, which is conducted at greater than 75% power. This note does not refer to any interim calibrations that are conducted below 75% power.

2. Bases Section B 3.2.4, QPTR, Surveillance Requirements section: The Bases for SR 3.2.4.1 are revised to modify the reference to Note 2. The qualifying statement, "if more than one input from Power Range Neutron Flux channels are inoperable," was deleted. This qualifier was not part of the actual Note 2 in SR 3.2.4.1. It was deleted from the standard version of this SR note as part of a generic change through the TSTF process. Due to an oversight, the Bases were not previously corrected.

3. Bases Section B 3.3.2, Applicable Safety Analyses, LCO, and Applicability section, page B 3.3.2-14: There are incorrect references to the number of switches used to actuate manual phase A and B and which switch actuates which train. This error had been previously corrected in other Bases sections, but these references were missed. They have now been corrected.

4. Bases Section B 3.4.14, RCS Pressure Isolation Valve (PIV) Leakage, Background and References sections: There was an incorrect reference to UFSAR Table 6-77 as the location of the list of PIVs. The PIVs are actually contained in new UFSAR Table 5-1. This reference was corrected.

5. Bases Section B 3.5.2, ECCS-Operating, Surveillance Requirements section: The Bases for SR 3.5.2.5 and SR 3.5.2.6 states, "These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal ..." Current TS requirement 4.5.2e.1 also includes a Containment Sump Recirculation signal for demonstration of this requirement. The affected ITS Bases section has been modified to include this signal as well (in addition to the SI signal).

6. Bases Sections B 3.6.8, HSS, Surveillance Requirements section and B 3.6.11, ARS, Surveillance Requirements section: These two Bases sections are revised to add information from the old TS Bases Section 3/4.6.5.6 (Containment Air Return and Hydrogen Skimmer Systems) which indicates that an air density correction must be made to the surveillance results to account for environmental conditions following an accident. This correction has already been made in plant procedures, so this change merely aligns

the Bases wording with current practices. Also, the Bases for SR 3.6.11.2 was editorially revised to delete the extraneous "to be".

7. Bases Section B 3.6.13, Ice Condenser Doors, Surveillance Requirements section: SR 3.6.13.7 inadvertently omitted the equality sign from item a. The entry should read <= 37.4 lb. This entry has been corrected.

8. Bases Section B 3.7.8, NSWS, LCO section: Information was added to indicate that the use of a NSWS pump and associated diesel generator on a shutdown unit to support continued operation of a unit beyond 72 hours with an inoperable NSWS pump is an unreviewed safety question. This information is consistent with current practice not to allow such an alignment.

9. Bases Section B 3.7.9, SNSWP, Applicable Safety Analyses section: The Bases was editorially revised to delete the reference to the 2.4 degree margin (penalty) described in the Catawba SER. On February 5, 1997, a letter was sent from P.S. Tam (NRC) to W.R. McCollum (Catawba), which informed Catawba that the 2.4 degree penalty no longer needed to be imposed and that the Bases of the Technical Specifications could be revised to delete reference to this penalty.

10.Bases Section B 3.7.12, ABFVES, Background section: The Bases were revised to remove those sentences which refer to the ABPVES as a standby system. The system is not presently being used as a standby system, since the fans are always running and the dampers are in their emergency realignment position. This configuration is being maintained until future implementation of a planned modification to restore the system to a standby system. Also, the discussion is revised to more accurately describe system function following an Engineered Safety Feature Actuation System signal.

11.Bases Section B 3.7.12, ABFVES, LCO section: The Bases were revised to add information that states that fan power supply is provided by buses which are shared between the two units. If normal or emergency power to the ABFVES becomes inoperable, then the LCO Required Actions must be entered for each unit in the applicable mode.

12.Bases Section B 3.8.4, DC Sources-Operating, Background section: The Bases were editorially revised to indicate that the subject discussion on ventilated rooms and being apart from its charger and distribution centers applies to the 125 V Vital DC batteries and not the 125 VDC DG batteries. Also, the information regarding selection of an available commercial battery is deleted.

13.Bases Section B 3.8.4, DC Sources-Operating, Surveillance Requirements section: The Bases for SR 3.8.4.7 was revised editorially to indicate that the specified values are minimum values. The Bases for SR 3.8.4.9 is revised to indicate that per IEEE-450, degradation is indicated when battery capacity drops by more than 10% relative to its average capacity on the previous performance tests.

14.Bases Section B 3.8.4, DC Sources-Operating, References section: Reference 9 was revised editorially (changed "and" to "and/or").

15.Bases Section B 3.8.5, DC Sources-Shutdown, Actions section: "A.1" was typographically corrected to "A.1.1".

16.Bases Sections B 3.8.7, Inverters-Operating, B 3.8.8, Inverters-Shutdown, B 3.8.9, Distribution Systems-Operating, and B 3.8.10, Distribution Systems-Shutdown, Surveillance Requirements section: The affected SR Bases were revised to add the word "indicated" as it pertains to verification of proper voltage.

17.Bases Section B 3.8.9, Distribution Systems-Operating, Table B 3.8.9-1: The word "NOMINAL" is added to the VOLTAGE column to indicate that these values are indeed nominal values.

Evaluation: While these Bases changes involve Bases pertaining to a number of plant structures, systems, and components (SSCs), in no case does the change result in any change to the plant or the way it is operated or maintained. No SSC is modified or caused to operate in a different manner from current operation. No seismic, environmental, materials, or reactivity effects are created on these SSCs as a result of the proposed Bases changes. No new failure modes/effects or new types of system/component interactions will be introduced upon any SSC as a result of these changes. SSC behavior during steady state and transient conditions will be unaffected by any of these Bases changes. Plant response to any external phenomena, natural or man-made, likewise will not be impacted. No plant transient 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 SSCs are being maintained. Finally, no actual regulatory commitments are being eliminated or reduced by these changes. There are no unreviewed safety questions associated with these changes. No Technical Specification Changes are required. No UFSAR changes are required.

150 **Type:** Miscellaneous Items **Unit:** 0

Title: Mk-BW BPRA Design Changes

Description: Two design changes are being made to the Mk-BW BPRA rod assemblies. One involves the spacer tube that is used to position the poison pellets and the other involves the drill out on the upper and lower end caps. Each change is explained further below.

These design changes were not implemented at the same time. The design change for the new tubular spacer was incorporated for Catawba 1 Cycle 12 (CIC12), Contract 134M. The changes to the end cap drill outs were not implemented on the CIC12 design.

Switch from Solid Spacer to Tubular Spacer

Both of the design changes being evaluated in this document involve minor changes to the Mk-BW BPRA rod. One of these changes involves the BPRA spacer. The absorber stack in the BPRA rod is made up of A1203-B4C pellets. These pellets are sealed within zirconium tubing and positioned within the rod by a solid zirconium spacer, which rests on top of the lower end cap. In an effort to standardize the spacer material used for their various BPRA products and to use a more economical part, Framatome Cogema Fuels (FCF) Company has developed a new spacer design for the BPRA rods. The main difference between the two spacers is that the new design uses a hollow or tubular spacer. This particular change does not impact any of the other parts that comprise the BPRA rod. The change was intentionally designed such that the spacers are the same height, which means that the relative location of the poison stack in the rod is unchanged in the BPRA using the new tubular spacer design. Another difference between the two designs is that the tubular spacers have a slightly smaller diameter. Because the new spacers are tubular and have a smaller outside diameter, the overall weight of the BPRA rod, and subsequently the BPRA assembly itself, will be slightly less.

The impact of these changes on the fuel performance have been evaluated and determined to be acceptable as documented in calculation DPC-1553.26-00-0153

Increased Drill Out in the Upper and Lower End Caps

The other design change that will be documented as a part of this evaluation is an adjustment to the BP upper and lower end cap drill outs. For the upper end caps, the design is being changed to add a drill out. For the lower end cap, the design is being changed to increase the drill out diameter. One purpose of these design changes is to permit use of the fuel rod welding equipment and subsequent inspection by the Andrex radiographic weld inspection equipment. This ultimately makes the BP end caps similar to the fuel rod end caps and enables BPRA fabrication on the fuel rod production line.

The impact of these changes on the fuel performance have been evaluated and determined to be acceptable as documented in DPC-1553.26-00-0153.

Evaluation: All Systems, Structures and Components that are involved or affected by the design changes being made to the BP spacer and end caps were identified and reviewed. Since all of these changes impact the BPRA rods, the only Systems, Structures or Components involved or affected are the BPRA themselves. The only design basis accidents that involve BPRAs are the core misloading, when a BPRA is located in the wrong fuel assembly, and reactivity insertion, when a BPRA is ejected from the core. Note though that the BPRA ejection is not considered a credible scenario. The design changes being evaluated here do not impact either of these accident scenarios in any way because the changes only affect the internal parts of the BPRA. The UFSAR updates will not affect the design bases events as the text change only modifies the description of the BPRAs.

The only industry failure mechanism for BP rods has been a seal weld failure. The consequence of this failure was a loss of poison from the BP and eventual unit shutdown. The BPRA seal weld is not affected by the introduction of the tubular spacer, the new end cap drill outs, or the text change to the UFSARs. Note though that the new drill outs on the end caps help to ensure that heat sink issues do not adversely affect the end cap weld quality during BPRA fabrication on the fuel rod line. Use of the Andrex radiographic weld inspection equipment, the same weld inspection equipment used for fuel rods, ensures that only welds that meet the weld inspection acceptance criteria are deployed as final product. No new failure modes are introduced as a result of implementing the design changes or the UFSAR text change. There is no unreviewed safety question associated with this UFSAR change. No Technical Specification chnages are required. A change is required for UFSAR Section 14.2.1.3.2 and 4.2.3.2.1

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

Title: Operator Aid Computer Alarm Responses ALM5062 and ALM9062 (Hotwell Discharge Temperature)

Description: Operator Aid Computer Alarm Responses ALM5062 for Unit 1 and ALM9062 for Unit 2 provide computer alarms to direct operator actions on Hi (130 degrees F) and Hi-Hi (132 degrees F) Hotwell Pump discharge temperature. The alarm response currently directs Operations to maintain hotwell pump discharge header temperature less than 130 degrees by reducing load if the "Hi" alarm is received. The alarm response for Hi-Hi currently directs Operations to declare the Auxiliary Feedwater System inoperable if the temperature is greater than or equal to 132 degrees F. The original alarm response was developed in response to Auxiliary Feedwater System suction temperature concerns due to an Upper Surge Tank overheating event. After the original alarm response was developed, valve CM-33 (Hotwell High Level Control valve) was identified as an adverse system interaction which affects operability of the Auxiliary Feedwater System. Because of this adverse system interaction valve CM-33 is mainatined isolated per a compensatory action. Since the compensatory action is in effect, the requirement for maintaining the hotwell pump discharge temperature less than 130 degrees is no longer required. The OAC alarm response will be changed to delete the requirement to maintain hotwell pump discharge temperature less than 130 degrees as long as the compensatory action to keep valve CM-33 closed is in effect.

Evaluation: This change to Operator Aid Computer Alarm Responses will allow the Condensate Makeup System to be operated at temperatures within its design limits. Condensate Makeup System operation is not credited in accident mitigation. Low condenser vacuum alarms provide the function of alerting operators of degrading vacuum conditions. There is no effect on accident initiation. The compensatory action in place to keep valve CM-33 closed is sufficient to prevent feeding the Auxiliary Feedwater System with high temperature condensate water. Although the Auxiliary Feedwater System can be fed directly from the hotwell, an analysis has shown that it is not possible to take suction from the hotwell with accident flows present. There are no unreviewed safety questions associated with this change. No Technical Specification changes are required. No UFSAR changes are required.

47 Type: Miscellaneous Items **Unit:** 0

Title: Revision to Selected Licensee Commitment 16.9.3, CO2 Systems

Description: This change to Selected Licensee Commitment 16.9.3 will revise the frequency for testing requirement (b)(i). This testing requirement currently requires verification of the C02 storage tank level to be greater than 44% for all Low Pressure (LP) C02 systems (Diesel Generator Rooms) at least once every 7 days. This Selected Licensee Commitment change will revise the frequency of this requirement to at least once every 31 days.

The tank level verification is performed to ensure, on a weekly basis, that an adequate quantity of C02 is available, as recommended by NFPA 12. The Low Pressure C02 system tank levels are checked every night on operator shift rounds. Operations is procedurally required to request filling the Low Pressure system C02 tanks anytime the tank levels fall below 90%. Review of the results of the Diesel Generator Weekly Test procedure shows that the tank level has never been found to be below 44% on a weekly basis since July of 1997. The intent of the requirements given in NFPA 12 is met by the performance of the nightly Turbine Building Rounds. The Selected Licensee Commitment Surveillance is intended to monitor the condition of the Low Pressure C02 tank refrigeration system, thus resulting in assurance that the tank level and pressure is adequately maintained. Therefore, it is concluded that a monthly check of the C02 tank levels versus a weekly check is adequate. This change will enable the frequency of the performance of procedures PT/1,2/A/4450/010B to be changed from weekly to monthly. This change will result in a saving of operator manpower currently required to perform these procedures weekly.

Evaluation: This Selected Licensee Commitment change will only revise the required frequency for verifying the LP C02 tank level. The probability of fire event/accident is not affected by changing surveillance frequencies. This change will not impact the ability of the LP C02 system to be used to effectively fight a fire in the Diesel Generator Rooms. Thus, a change to the surveillance frequencies will have no impact on the consequences of an accident evaluated in the SAR.

No new accident types are created by this change to the Selected Licensee Commitment. Fire events are already postulated in the SAR. A change to the surveillance frequency will not result in the creation of any different accident events. The ability to maintain adequate assurance that the LP C02 system supply is acceptable is not changed. The combination of nightly Turbine Building rounds and a monthly surveillance is adequate to maintain verification of the C02 tank level. The probability of plant equipment malfunction is not affected. This Selected Licensee Commitment change will not affect the Catawba fire protection "defense in depth" philosophy of providing early detection and effective suppression. The C02 tank level is maintained well above the required 44% level by existing operational controls. The consequences of a fire in any of the D/G rooms are not affected by this change. The SAR already postulates malfunctions due to fire events. The SLC change described in this package does not add any new failure modes other than those already postulated in the SAR. The SLC change described in this package has no impact on the margin of safety of any nuclear safety related systems structures or components. Fire protection commitments are not contained in the Technical Specifications.

No Unreviewed Safety Questions are created by the Selected Licensee Commitment change described herein. No Technical Specification changes are required. No UFSAR Changes are required.

134 Type: Miscellaneous Items **Unit:** 0

Title: Revision to Technical Specification Bases B 3.7.10 associated with PIP 0-C98-4190

Description: The Technical Specification Bases (B 3.7.10) for the Control Room Area Ventilation System is being revised. The revision resulted from a problem in which the guidance associated with closing a Control Room Area Ventilation System outside air intake was ambiguous. The change only clarifies the intent of the original bases wording and does not make any technical or procedural changes.

Evaluation: One of the design bases of the Control Room Area Ventilation System is to be able to pressurize the control room to prevent the entry of dust, dirt, smoke, toxic gases, and radioactivity. The proposed change to the Technical Specification bases does not impact this design basis. The change is being made to clarify that an operator has an option of either leaving an outside air intake open or closing it in the event of a smoke or radiation alarm. The Control Room Area Ventilation System outside air intakes contains smoke, radiation and chlorine detectors. While the chlorine detectors cause an automatic closure of an affected intake, the smoke and radiation detectors only provide control room alarms. It is then up to the operators to determine if the intake should be closed or left open. This bases clarification is made to provide additional information as to why an intake may be closed or why it may be acceptable to leave it open. Closing an intake is acceptable because the Control Room Area Ventilation System can pressurize the control room with only one of the two intakes open. Leaving the intake open is acceptable because the control room pressurizing filter trains are designed to remove smoke and radiation contaminants prior to supplying air to the control room. Thus, with an intake closed or open the Control Room Area Ventilation System will perform its design basis function. The change being made does not alter the way the Control Room Area Ventilation System is operated or change any procedural guidance. It merely clarifies the intent and wording of the existing Technical Specifications.

There are no unreviewed safety questions associated with this change to the bases for Technical Specification 3.7.10. There will be no modification in the way the operators respond if an outside air intake closes based on this Technical Specification basis change. No accidents previously analyzed in the SAR are impacted by this change. All components of the Control Room Ventilation System, including the outside air intake chlorine monitors, will be fully operable and are unaffected by this change. No Technical Specification changes are required. No UFSAR changes are required.

46 **Type:** Miscellaneous Items **Unit:** 0

Title: Revision to Technical Specification Bases Section for SR 3.6.6.5 and 3.6.6.6

Description: The following changes are being made to bases for Surveillance Requirements (SR) 3.6.6.5 and 3.6.6.6:

Currently, the surveillance bases states: "..each containment spray pump discharge valve opens or is prevented from opening and each containment spray pump starts or is de-energized and prevented from starting upon receipt of the Containment Pressure Control System (CPCS) start and terminate signals."

Surveillance Requirement 3.6.6.6 states: "..each spray pump discharge valve closes or is prevented from opening upon receipt of a terminate signal and is allowed to open upon receipt of a start permissive....".

The revised bases text will read:

"These Surveillance Requirements require verification that each containment spray pump discharge valve closes or is prevented from opening upon receipt of a CPCS terminate signal and is allowed to open upon receipt of a CPCS start permissive. In addition, it must be shown that each spray pump is allowed to start or is de-energized and prevented from starting upon receipt of CPCS start and terminate signals."

CPCS is the Containment Pressure Control System.

Evaluation: There is no unreviewed safety question involved with this change. No Technical Specification changes are required. No UFSAR changes are required.

This change is editorial. The revised Surveillance Requirement bases section does not allow actions that are more restrictive or less restrictive than the requirements already presented in SR 3.6.6.5 and SR 3.6.6.6. Operability of the Containment Spray system will still be based on the surveillance requirements. The sole purpose of the change is to remove confusion from the bases section by making the text agree with the wording of the Surveillance Requirements. Neither the Containment Spray System nor the CPCS is defined in the SAR as an accident initiator. This change is editorial and does not affect any of the surveillance requirements that define operability for these systems. The change will not allow actions that are more restrictive or less restrictive than the requirements already presented in Surveillance Requirements 3.6.6.5 and SR 3.6.6.6. This change merely makes the bases text consistent with the wording from the Surveillance Requirements. This change does not result in an increase in offsite dose for any accident previously evaluated. The change does not affect any of the fission product barriers. This change will not allow performance of unanalyzed activities that may cause an accident different than what is currently evaluated. The Surveillance Requirements Bases section only serves to clarify the requirements presented in the actual Surveillance Requirements 3.6.6.5 and 3.6.6.6 which are not being revised. No new failure modes are created by this editorial change to the Tech Spec bases. The change will not allow actions that are more restrictive or less restrictive than the requirements already presented in Surveillance Requirements 3.6.6.5 and 3.6.6.6. No fission product barriers will be adversely affected by this change, nor will the change affect the current operation of these systems. Operability of these systems will still be based on the current Surveillance Requirements which have not changed.

257 Type: Miscellaneous Items **Unit:** 0

Title: Revision to the Bases for Technical Specification 3.6.5

Description: The Bases for Technical Specification 3.6.5 is being revised to show that the minimum initial lower containment temperature is 95 degrees F. versus the present value of 100 degrees F. Clarifications will be added to address when and why instrument uncertainty is added to temperature values obtained from analyses and used in the associated Technical Specifications

Evaluation: The current Tech Spec LCO and Bases temperatures for minimum and maximum temperatures in the upper and lower containments are as follows:

Tech Spec 3.6.5 LCO (Mode l) temperature values (Amendment 173/165) are as follows:

Upper Containment Deg F.	Lower Containment Deg F.
75 min	100 min
100 max	120 max

Tech Spec 3.6.5 describes the bases for the following temperature values:

Upper Containment Deg F.	Lower Containment Deg F.
70 min (due to LOCA)	100 min (due to LOCA)
100 max (due to SLB)	135 max (due to SLB)

It was determined that the 100 degree minimum cited in the Technical Specification Bases for lower containment should be changed to 95 degrees based on the initial temperatures used in calculation CNC-1552.08-00-0278 Rev 0, "FSAR Section 6.2-1: Containment Response Analysis for Ice Weight Reduction Dated 1/21/98". The analysis shows that for the present configuration the containment can meet the pressure transient requirements with a reduced initial temperature of 95 degrees. The ice weight reduction analysis looks at containment temperature and pressure responses due to LOCA.

For the peak containment pressure analysis (LOCA) the minimum temperatures for upper and lower containment in the LCO reflect a 5 degree difference between the calculation value and the surveillance value. The instrument loop inaccuracy is determined in calculation CNC 1210.04-00-0054 "Upper and Lower Containment Temperature (Containment Ventilation System) Instrument Loop Accuracy". According to the calculation, the lower instrument loops with a span of 40-400 deg F have an uncertainty of up to +/- 6 degrees and loops with a span of 40-200 deg F have an uncertainty of up to +/- 5.3 degrees. The Operator Aid Computer temperature readings factor in the effect of averaging on the uncertainty. The averaging effect tends to reduce the overall magnitude of the uncertainty. With the limiting case of two lower containment units running, and assuming one of the four RTD's as a single failure, three Operator Aid Computer input temperature readings would be averaged to determine instrument error. The calculation inaccuracy is calculated as [(6^2 + 6^2 + 5.3 ^2)/3^2]^.5 = 3.3 degrees. The 5 degree difference used is therefore conservative for the lower containment lower temperature limit. For the upper containment lower limit, one upper containment ventilation unit is the limiting case for operation in Mode 1. Per calculation CNC 1210.04-000054, the maximum error is +/- 2.7 degrees F. The 5 degree difference used is therefore conservative for the upper containment lower temperature limit.

For the maximum upper containment temperatures the LCO limit is 100 degrees. This temperature is the same as the Bases temperature and is the initial temperature used in the Steam Line Break Containment Response Analysis . The 100 degrees value does not require an instrument inaccuracy adjustment for use in the LCO since the 100 degree value is not considered significant because the Steam Line Break DBA is sensitive to the lower containment initial temperatures but not the upper containment initial temperatures.

The maximum lower containment temperature of 120 degrees is assumed in the Steam Line Break Containment Response Analysis. The calculation which investigates the Steam Line Break is run not only at 120 degrees but also at 125 and 135 degrees as sensitivity cases. The Tech Spec Bases reflects the sensitivity case run with 135 degrees. At this initial temperature, the peak lower containment temperature reaches 317 degrees F., which is within the environmental qualification limit for long term operation.

The Tech Spec Bases only discuss the minimum and maximum temperature limitations for the most limiting analyses. The Tech Spec Bases will be revised to reflect the 95 versus 100 degree lower containment temperature. Also, the Bases will be revised to provide justification as to why instrument accuracy is not required for the 100 degree F. Upper Containment Max temperature, and why 135 degrees F. is used as the maximum lower containment temperature.

There is no unreviewed safety question associated with this bases change. The Containment Ventilation System is not an accident initiator in any accident scenarios in the UFSAR. The change to the Bases clarifies the temperature limits used in the LCO and provides an explanation of the conservatism built into the temperatures. No Technical Specification changes are required. No UFSAR changes are required.

149 **Type:** Miscellaneous Items **Unit:** 0

Title: Revisions to Bases for Technical Specification 3.5.1, 3.5.4, 3.6.12 and revisions to the Equipment Qualification Criteria Manual

Description: The activity revises the bases for Technical Specification 3.5.1 (Cold Leg Accumulators - CLA's), TS 3.5.4 (Refueling Water Storage Tank - FWST), and TS 3.6.12 (Ice Condenser). Combined, the activity associates the requirements for boron concentration in the solution in these structures to an allowable range of 7.5 - 9.3 for containment sump pH following the design basis LOCA. The activity also revises the Environmental Qualification Criteria Manual (EQCM), Table 8.0-1, Note 3 to change the limits of spray pH. Currently given as 4.0 - 9.0, the revised range in the EQCM is 4.0 - 9.3 during injection and 7.5 - 9.3 afterwards. The activity also replaces a specific value of spray boron concentration (2100 PPM) with a reference to the Core Operating Limits Report (COLR).

Evaluation: This activity defines a margin of safety currently not in the basis for the Technical Specifications. Prior to conversion to the Improved Technical Specifications (ITS), a range for post LOCA containment sump pH had been specified in the basis for the Technical Specifications for the Cold Leg Accumulators and the Refueling Water Storage Tank. This range was 7.5 - 9.5. Therefore, the range proposed for the bases of TS 3.5. 1, TS 3.5.4, and TS 3.6.12 adds margin relative to the range in the Technical Specification bases before conversion to the ITS. The changes to the lower limit of pH for chemical spray in the EQCM adds margin. The changes to the upper limit does not adversely affect the qualification of safety related equipment in containment as their specifications included an upper limit of 10.5 for spray pH.

No unreviewed safety question is associated with the activity. No change to any Technical Specification is required. No change to the UFSAR is required.

43 Type: Miscellaneous Items **Unit:** 0

Title: Selected Licensee Commitment 16.5.2, Reactor Coolant System - Safety Valves

Description: When Improved Tech Specs (ITS) were implemented at Catawba, the previous Catawba Technical Specification applicable for Pressurizer Safety Valves (PSVs) during Modes 4 and 5 (TS 3.4.2.1) was carried over to the Selected Licensee Commitment (SLC) Manual with no clarifications to account for the 4.5 square inch vent path allowed by ITS 3.4.12, Low Temperature Overpressure Protection (LTOP) System. A change to SLC 16.5-2, Pressurizer Safety Valves (Mode 4 at less than 285 °F, and Mode 5) was implemented to clarify the associated Catawba Licensing Commitment.

The PSVs do not perform an adequate Low Temperature Overpressure protection (LTOP) function in the stated modes of applicability. ITS 3.4.12 assures that the PORVs are operable to perform this LTOP function. A PSV that is removed from its flange meets the LTOP option of creating a Reactor Coolant System vent of greater than 4.5 square inches per ITS LCO 3.4.12 b. During cold shutdown and draining of the Reactor Coolant System, there is no adverse effect of opening more than one 4.5 square inch path. Therefore, a minimum of one PSV that is removed provides an LTOP path that is equivalent and superior to a minimum of one PSV that is operable in response to a mass input or heat input transient. The Remedial Action to immediately suspend all operations involving positive reactivity changes is conservative to preclude any reactivity management event. The Remedial Action to place an OPERABLE residual heat removal loop into operation in the shutdown cooling mode makes the residual heat removal system suction relief valve available to provide its equivalent LTOP function.

Evaluation: No Technical Specifications changes are required. There are no Unreviewed Safety Questions associated with this change. 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 and in discussions regarding Low Temperature Overpressure Protection in sections 5.2.2, 7.6.20, and 6.3.2.5. The fission product barriers of the pellet, clad, the reactor coolant system pressure boundary and containment are not affected as a result of this change to the SLC manual.

67 **Type:** Miscellaneous Items **Unit:** 0

Title: Selected Licensee Commitment 16.9.5

Description: The testing requirement of Selected Licensee Commitment (SLC) 16.9.5(b) requires the inspection of the closing mechanism and the latches for each fire door listed in SLC Table 16.9-4. The design basis of the Turbine Driven Auxiliary Feedwater Pump Pit Doors (AX217F and AX260F) is such that they were intentionally designed without a closure device or latching mechanism (reference drawings CNM-1182.00-0100 and CNM-1182.00-0142). This design approach was used based on the need for rapid egress from the pit due to the CO2 system and in order to provide pressure relief during a CO2 discharge.

Evaluation: This change clarifies the requirement for inspecting these doors in accordance with their design basis. The probability or consequences of a fire event are not affected by this clarification. There are no unreviewed safety questions associated with this change to SLC 16.9.5. No Technical Specification changes are required. No UFSAR changes are required.

241 **Type:** Miscellaneous Items **Unit:** 1

Title: Temporary Modification CNTM-0007

Description: Temporary Modification CNTM-0007 will have the following effects:

1. Disable the input from the Reactor Coolant Pump 1A lower bearing temperature measurement to Control Room Annunciator 1AD-7 B/2.
2. Change the associated Operator Aid Computer point from 170 degrees F. to 175 degrees F.
3. Change the dead band associted with the Operator Aid Computer point from 2 degrees F. to 0.5 degrees F.

These changes will remove nuisance alarms associated with identified noise spikes on the measurement loop.

Evaluation: There are no unreviewed safety questions associated with this temporary modification. There are adequate alternate means for monitoring Reactor Coolant Pump Motor 1A using the Control Room Annunciator on seal water outlet temperature and pump vibration. No Technical Specification changes are required. No UFSAR changes are required.

86 **Type:** Miscellaneous Items **Unit:** 1

Title: Temporary Station Modification Work Order 9814849601/(02)

Description: Temporary Station Modification (TSM) Work Order 9814849601/(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/l1/A/6100/002 Rev 137B has been revised to include provisions for Mode 5 boron concentration at 200 degrees F 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 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 290 - 300 degrees F) . 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 9814849601/(02) and associated procedure OP/l/A/6100/002 Rev 137B . 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 used, a final determination on updating the UFSAR 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.

204 **Type:** Miscellaneous Items **Unit:** 1

Title: Temporary Station Modification Work Order 98209089 to gag valve 1RC31

Description: Temporary Station Modification Work Order 98209089 will install a gag on valve 1RC31. The work order will repair a problem on the operator for valve 1RC31, "Cooling Tower 1A Make-up Water Flow Control Valve". While the operator is being repaired, the make-up flow path through valve 1RC31 will need to be inservice. This temporary modification will install a gag on valve 1RC31 to main the valve in the full open position while the operator is removed and make-up flow to the 1A Cooling Tower is still in service. Chemistry personnel will control flow using the manual butterfly valve 1RC95 upsteam of valve 1RC31 and procedure OP/0/B/6400/017.

While this temporary modification is installed, make-up to the 1B Cooling Tower will be in normal service with automatic makeup flow control provided by valve 1RC32.

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

254 **Type:** Miscellaneous Items **Unit:** 0

Title: Use of Westinghouse RFA Fuel/Burnup Increase for Mark-BW Fuel

Description: The following design burnup changes are being evaluated for Westinghouse RFA Fuel and Mark-BW Fuel:

Westinghouse Fuel Assembly Pellet Burnup Limit previously "N/A" revised to "N/A".
Westinghouse Fuel Assembly Pin Burnup Limit previously "62 GWD/t" revised to "62 GWD/t".
Westinghouse Fuel Assembly Assembly Burnup Limit previously "N/A" revised to "N/A".

Mark-BW Fuel Assembly Pellet Burnup Limit previously "66 GWD/t" revised to "N/A".
Mark-BW Fuel Assembly Pin Burnup Limit previously "60 GWD/t" revised to "60 GWD/t".
Mark-BW Fuel Assembly Assembly Burnup Limit previously "55 GWD/t" revised to "N/A".

There are also other slight dimensional and material changes to various components of the fuel assembly (i.e., the Westinghouse RFA fuel as compared to Mark-BW fuel), including but not limited to:

Cladding material (Zircalloy-4 to ZIRLO). ZIRLO is a combination of ZIRC2 and 1% (by weight) niobium (Nb).

Dry fuel assembly weight (1470 lbm to 1454 lbm) and wet fuel assembly weight (1315 lbrn to 1269 Ibm).

The existence of a bottom (debris) protective end grid.

Fuel rod ID (0.326 inches to 0.329 inches).

Overall rod length (151.797 inches to 152.6 inches).

The use of IFBA for selected rods.

Pellet OD (0.3195 inches to 0.3225 inches).

Pellet length (0.4 inches to 0.387 inches).

Pellet percent of theoretical density (96 to 95.5).

The scope of this evaluation is limited to the effects of the change on:

Offsite and control room operator dose calculations.
Post-accident shielding calculations.
Spent Fuel Pool decay heat, temperature and heat removal assumptions and calculations.
Post-accident sump pH calculations.

Evaluation: Since the change in allowable fuel burnup limits is not significant, based on engineering judgment, there will be no substantial change in dose results to offsite points (EAB or LPZ) or the control room operator.

Since the change in allowable fuel burnup limits is not significant, based on engineering judgment, there will be no substantial change in results of post-accident equipment or inplant personnel dose (for post-accident in-plant accessibility pursuant to NUREG-0737).

Since the change in allowable fuel burnup limits is not significant, based on engineering judgment, there will be no substantial change in the results to Spent Fuel Pool (SFP) decay heat calculations or SFP temperature. Additionally, per the Total Core Unloading procedure (step 8.16), SFP decay heat calculations are routinely being performed prior to core unloading .

It has been determined that the boron in the coating on the IFBA fuel will have an insignificant effect on post-accident sump pH.

The response to a "Request for Additional Information regarding the Design Basis Accident, Dropping of Weir Gate onto Spent Fuel" contains very specific commitments regarding fuel temperature and the methodology used to compute those values (and the effects on gap fraction), and fuel pin pressure and the resultant effects on pool decontamination factor in a postulated Fuel Handling Accident (FHA). Use of Westinghouse fuel will necessitate changes to the commitments made concerning the FHA. This safety evaluation will not address these issues. The commitments will be handled by a License Amendment Request (LAR) prior to unloading of the Westinghouse fuel to the Spent Fuel Pool.

Fuel assembly displacement is utilized in the design basis Spent Fuel Pool heatup calculations. Specifically, the pool volume modeled to be heated in a loss of forced cooling event (Standby Shutdown Facility event) is reduced by the volume of the fuel cells, racks, and assemblies. The fuel cells and racks are not affected as a result of this change. The RFA fuel assembly displacement is 4613.76 in 3 (2.67 ft3). Calculation CNC-1201.30-00-0014 utilized a fuel assembly displacement of 2.6545 ft^3 for the heatup calculations. Hence, this change causes a slight reduction in the volume of water in the Spent Fuel Pool (heat sink for the fuel decay heat). This change is (assuming that all current and future assemblies in the Catawba SFP are RFA assemblies) 0.024%.

This change is insignificant, and would not be displayed in the precision of the heatup calculation.

Calculations CNC-1140.00-01-0001 ("Spent Fuel Storage Racks") and CNC- 1140.00-020001 ("New Fuel Storage Racks Analysis and Design") address seismic effects on the fuel assembly racks. Since the weight of the Westinghouse fuel is less than the weight assumed in the structural analysis, this change will not have an adverse effect on the structural analysis.

There are no unreviewed safety questions associated with this change. None of the issues addressed are accident initiators. No Technical Specification changes are required. No UFSAR changes are required.

141 **Type:** Miscellaneous Items **Unit:** 2

Title: Work Order 98168459, Use of Polar Crane in Mode 4 to replace Hydrogen Igniter Glow Plugs in the Hydrogen Mitigation System

Description: Maintenance personnel will replace one of the glow plugs associated with the Unit 2 Hydrogen Mitigation System, Train A located in the Unit 2 Containment. This will require the use of the Polar Crane. The Polar Crane will be used to position a maintenance worker under the glow plug to be replaced. Unit 2 will be in Mode 4 when this task is performed. The procedure for operation of the Polar Crane authorizes the removal of Polar Crane restraints (a pre-requisite for operation) between Mode 1 and Mode 4 "for the purpose of preventative maintenance of the Polar Crane, removing hatch covers and material handling for pre-approved modifications." The procedure does not literally allow operation of the Polar Crane for a maintenance activity. Replacement of a glow plug is considered a maintenance activity. Therefore the procedure will be revised so that restraints can be removed any time between Mode 1 and Mode 4 for management approved maintenance activities critical to operation with a 10CFR50.59 evaluation.

Evaluation: There are no unreviewed safety questions associated with this activity. The crane operators will follow a safe load path that was developed for movement of the Polar Crane while the Unit is between Mode 1 and Mode 4. The breaker for the main hook will be open during this operation. This will preclude the possibility of a load drop on any system, structure or component important to nuclear safety and also ensure compliance with NUREG 0612 (Phase 1). No Technical Specification changes are required. No UFSAR changes are required.

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

Title: Work per Procedure IP/0/A/3890/001 and Work Order 98155832

Description: This activity consisted of removing two thermometers from thermowells in the Control Room Chilled Water System and installing calibrated gauges in their place. The work was guided by Maintenance troubleshooting procedure IP/0/A/3890/01. This procedure will have steps included in it to remove the thermometers and install test instruments and also to replace the thermometers at the end of the evolution. This work was performed to support HVAC technicains in troubleshooting chiller oil sump temperature problems.

Evaluation: The Control Room Chilled Water System chillers are designed to provide chilled water to the Control Room, the Control Room area, and the Switchgear air handing units. These air handling units ensure that the ambient air temperatures in these areas do not exceed the allowable temperatures for personnel, equipment and instrumentation. The thermometers that are being removed from the system measure the chilled water inlet and outlet temperatures on the chiller. These thermometers provide local indication only and are not used in any accident scenarios. The thermometers are shown on the system flow diagrams and thus a procedure is needed to remove them. Removing the thermometers will not affect the operability or reliablity of the system. The thermometers are not a part of the system pressure boundary. This activilty will not have any effect on the probability or consequences of accidents analyzed in the UFSAR. There are no Unreviewed Safety Questions associated with this activity. No Technical Specification changes are required. No UFSAR changes are required.

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

Title: Work per Procedure IP/0/A/3890/01 and W/O 98149227 concerning the Control Room Ventilation Chiller System System instrumentation)

Description: Procedure IP/0/A/3890/01 and W/O 98149227 address removal of two thermometers from thermowells in the Control Room Ventilation Chiller System. The two thermometers (0YCTH5900 and 0YCTH5970) will be temporarily removed and calibrated test gauges will be installed in their place. The work will be controlled by Instrument and Electrical Troubleshooting Procedure IP/0/A/3890/01. This procedure will contain steps addressing removal of the thermometers , installation of the test instrumentation, and reinstallation of the termometers at the end of the evolution. This activity is being performed to support HVAC technicians in troubleshooting chiller oil sump temperature problems.

Evaluation: There is no unreviewed safety question associated with this activity. The thermometers provide local indication of chilled water inlet and outlet temperatures. They are not used in any accident scenario. The thermometers are shown on the system flow diagram and per station policy a procedure is required for their removal. Removing the thermometers does not affect the operability or reliability of the system. The thermometers are located in thermowells and therefore are not a part of the system pressure boundary. No Technical Specification changes are required. No UFSAR changes are required.

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

Title: Work per Procedure IP/0/A/3890/01 and W/O 98157524 (concerning the Control Room Ventilation Chiller System System instrumentation)

Description: Procedure IP/0/A/3890/01 and W/O 98157524 address removal of a thermometers from a thermowell in the Control Room Ventilation Chiller System. The thermometer (0YCTH5900) will be temporarily removed and calibrated test gauge will be installed in its place. The work will be controlled by Instrument and Electrical Troubleshooting Procedure IP/0/A/3890/01. This procedure will contain steps addressing removal of the thermometers, installation of the test instrumentation, and reinstallation of the thermometers at the end of the evolution. This activity is being performed to support HVAC technicians in calibrating the chilled water temperature.

Evaluation: There is no unreviewed safety question associated with this activity. The thermometer provides local indication of chilled water outlet temperature. It is not used in any accident scenario. The thermometer is shown on the system flow diagram and per station policy a procedure is required for their removal. Removing the thermometer does not affect the operability or reliability of the system. The thermometer is located in a thermowell and therefore is not a part of the system pressure boundary. No Technical Specification changes are required. No UFSAR changes are required.

225 **Type:** Nuclear Station Modification **Unit:** 1

Title: Nuclear Station Modification CN-11190/00, Replace Power Supplies and Circuit Breakers in Radiation Monitor Cabinets

Description: Nuclear Station Modification CN-11190/00 will replace the DC Power Supplies for Radiation Monitor Cabinets with new DC Power Supplies. Also all 2.5 Amp circuit breakers will be replaced with 5.0 Amp breakers. Currently, each radiation monitor channel uses an individual 24 Volt Linear DC power supply which is inefficient. This causes an increased temperature in the cabinets which could adversely affect the operation of the monitors.

Evaluation: There are no unreviewed safety questions associated with this modification. The radiation monitors have no effect on accidents analyzed in the UFSAR. No new accident scenarios are created. No seismic concerns are created by the changeout of these power supplies. This modification will not increase the load on the power supplies. There are no Appendix R concerns because no cables are being routed. No Technical Specification changes are required. No UFSAR changes are required.

39 Type: Nuclear Station Modification **Unit:** 1

Title: Nuclear Station Modification CN-11379 Replace 6.9 KV Switchgear Tie Breakers with Faster Vacuum Breakers

Description: Nuclear Station Modification CN-11379 Revision 0 will replace the current 6900 Volt tie breakers with faster vacuum breakers, because the current fast transfer schemes have the potential to subject the switchgear loads (motors) to excessive restart torque. The modification will also install switches to provide the function of defeating the automatic fast transfer when the units are off line.

The Unit Main Power System is part of the Onsite Power System which directly interfaces with the Offsite Power System. The balance of the Onsite Power System consists of the Diesel Generators, batteries, controls and auxiliary power system. The Unit Main Power System includes the main generator, isolated phase busses, Generator Power Circuit Breakers and associated motor operated disconnects, main step-up transformers, four unit auxiliary (20.9/6.9 KV) transformers and one auxiliary (20.9/13.8 KV) transformer. The Unit Main Power System starts with the main generator which feeds two trains (A and B) of transformers, breakers and conductors. The main generator feeds: two "20.9 KV to 230 KV half sized" unit step up transformers, and four "20.9 KV to 6.9 KV half sized" unit step down auxiliary transformers, and one auxiliary "20.9 KV to 13.8 KV step down" transformer through the isolated phase busses and two Generator Power Circuit Breakers. The Offsite Power System consists of the entire station switchyard including Power Circuit Breakers, associated Motor Operated Disconnects, conductors, and protective relaying.

Power is fed from the Onsite Power System to the 6900 Volt Normal Auxiliary Power System. The 6900 Volt Normal Auxiliary Power System includes four switchgear assemblies of a split bus design including a short leg split from a long leg on each bus. The long and short legs of a respective switchgear are normally powered from opposite trains of the Onsite Power System. These switchgear assemblies are named 1TA, 1TB, 1TC, and 1TD. Normal power for these switchgear is supplied from 20.9/6.9 KV transformers 1T1A, 1T2A, 1T2B, and 1T1B. In order of short leg then long leg, the power sources on 1TA, 1TB, 1TC, 1TD are trains AB, BA, AB, BA.

The 6900 Volt tie breakers provide the capability to tie together the short and long leg of a particular switchgear assembly such that both legs are powered by the same train of the Onsite Power System. The 6900 Volt Normal Auxiliary Power System feeds the 4160 Volt Essential Auxiliary Power System. Normal alignment has train A 20.9/6.9 KV transformer 1T2A feeding switchgear 1TA (short leg) which feeds Essential bus 1ETA through 6.9/4.16 KV transformer 1ATC; and train B 20.9/6.9 KV transformer 1T1B feeding switchgear 1TD (short leg) which feeds Essential Bus 1ETB through 6.9/4.16 KV transformer 1ATD.

It is through these 6900 Volt tie breakers that the 4160 Volt Essential busses can be supplied from the opposite 6900 Volt train. A "Zone Lockout" is protective action provided in the switchyard that isolates one side of the Offsite Power System including the generator from a fault by opening the switchyard and associated Generator Power Circuit Breakers. Following this isolation, the tie breakers can power the de-energized

6900 Volt switchgear from the opposite train of Unit Main Power System with the resulting alignment having one 6900 Volt switchgear completely supplied from one auxiliary transformer. This automatic transfer will also occur for any reason an under voltage condition exists on a particular short or long leg provided the under voltage is not due to a fault on the bus.

The subject tie-breakers are non-safety related devices. Essential Power is provided to the safety related 4160 volt Essential Auxiliary Power System via the Class 1E Diesel Generators. Therefore, the normal power source to the station auxiliaries is mainly important to the extent that plant transients are not created through interruption of power. Interruption of power through the normal feed paths discussed above (1ATC and 1ATD), would create a blackout condition such that the load sequencer would start and load the Diesel Generator onto the affected Essential Bus. Additionally, interruption of power would terminate powered operation of the reactor coolant pumps and they would begin a coastdown due to the rotating inertia of the flywheel. Loss of forced flow in one loop will cause a Reactor Trip from an initial power level exceeding 48 % (P8). Below 48 % but greater than 10 % (P10), two loops must lose forced flow to result in a Reactor Trip. Per Tech Spec Table 3.3.1-1 and UFSAR Table 7-1, a 2 out of 4 undervoltage condition on reactor coolant pump motor voltage, will cause a Reactor Trip at less than or equal to 77 % normal voltage per UFSAR Section 7.2.1.1.2. This protection is provided by the Reactor Coolant Pump Monitor System. Unsuccessful transfer of power will result in a loss of flow. A slow transfer of power will result in a Reactor Trip. A Reactor Trip causes a Turbine Trip from any power level if the Turbine is not tripped. Thus, the 6900 Volt Normal Auxiliary Power System is identified as an accident initiator. The Unit Main Power System is also an accident initiator since its transformers feed the 6900 Volt switchgear.

Additionally, through auto-starting the Diesel Generators, challenges to accident mitigation equipment are increased if normal power equipment is made less reliable. Thus, the tie breakers are evaluated to be equipment important to safety as they can initiate a transient (Reactor Trip/ Turbine Trip) and cause challenges to accident mitigation equipment (Diesel Generators) if they fail to operate as designed when called upon. In addition to the above equipment other equipment such as Hotwell Pumps, Condensate Booster Pumps, and Condenser Circulating Water Pumps are powered from the 6900 Volt switchgear.

Evaluation: There are no unreviewed safety questions associated with this modification. All scenarios involving possible motor restart torque potential are improved by this modification due to the faster transfer time provided by the replacement breakers. The effects of this modification on breaker coordination has been evaluated. It was determined that the effects, if any, would be positive. No Technical Specification changes are required. A change is required for UFSAR Section 8.3.1.1.1.3.

3 **Type:** Nuclear Station Modification **Unit:** 1

Title: Nuclear Station Modification CN-11387/00, Install Bonnet vents on Valves 1NI-10B and 1NI-136B to eliminate pressure locking concerns

Description: Nuclear Station Modification CN-11387/00 installs bonnet vents on valves 1NI-10B and 1NI-136B to eliminate pressure locking concerns. Due to pressure locking concerns identified during the response to NRC Generic Letter 95-07, bonnet vents have been determined to be necessary to maintain long term operability on specified valves. Valve pressure locking occurs when high pressure fluid is trapped in the bonnet of a closed gate valve. The bonnet on such a valve could become pressurized to the point where the valve may not open if required to perform the intended safety function.
A bonnet vent relief path on each valve will be installed with a small globe valve in the line to isolate the path if necessary. The valves affected by this modification are 1NI136B (Residual Heat Removal Pump to Centrifugal Charging Pump Isolation Valve) and 1NI-10B (Centrifugal Charging Pump to Cold Leg Injection Isolation Valve).

Evaluation: These valves are flexible wedge valves with two seating surfaces. The bonnet vents will bypass one seat and relieve/equalize pressure between the process pipe and the bonnet. The vent relief path will be a one half inch stainless steel, nuclear safety related, Duke Class B line. It will meet temperature and pressure requirements of the interfacing system. The bonnet vent isolation valves provided will be globe valves and will be locked open. If a motor operated valve develops a seat leak, the globe valves can be closed. The applications associated with these valves will vent to the high pressure side of the valve. There will be be no changes to the motor operated valve or the associated system. The addition of the weight of the new components will be evaluated for impact to the stress analysis and support/restraints. No electrical considerations are involved with this modification. Material compatibility reqirements of UFSAR 6.3.2.4 are satisfied with the specified piping and valve materials. Both the new bonnet vent and interfacing flow paths are stainless steel. All of the modified valves are active valves described in UFSAR Table 3-104. Emergency Core Cooling System (ECCS) leakage should not be adversely affected with the addition of the bonnet vent isolation valves which are of packless design. This Nuclear Station Modification does not involve an Unreviewed Safety Question. No Technical Specification changes are required. Changes are required for UFSAR Figures 6-128 and 6-130 (piping flow drawings).

33 Type: Nuclear Station Modification **Unit:** 1

Title: Nuclear Station Modification CN-11388/00

Description: Nuclear Station Modification CN-11388/00 replaces the existing 125 VDC Diesel Generator Auxiliary Power Batteries. The replacement batteries will have 40 ampere-hours of additional capacity compared to the existing batteries. The replacement batteries will have two additional cells per battery for improved capacity. The new batteries will also have nickel plated negative plates to retard migration of iron at elevated temperatures for better reliability.

Also cooling units 1VDAH2A and 1VDAH2B will be removed along with the associated ducting and power. Air temperature thermometers 1VDTH5240, and 1VDTH5250 will removed. These thermometers currently exist mounted to the wall above the battery racks. An interference exists mounted to the wall above the battery racks. An interference exists since the new battery cells are taller than the current cells.

The 125 VDC Diesel Generator Auxiliary Power System provides a separate and independent train of 125 volt DC power to each diesel generator. Each train consists of 125 VDC battery and battery charger powered from its associated train of 600 volt essential power.

The diesel generator batteries are Nickel-Cadmium Class 1E batteries. These batteries are sized to carry their assigned loads for two hours.

Evaluation: This system provides direct support to ensure the proper operation of the emergency diesel genertor. Each Train of the 4160 Volt Essential Power system is supplied with emergency standby power from an independent diesel generator. Each diesel generator is designed to attain rated voltage and frequency and to accept load within 11 seconds after receipt of a start signal. This modification provides the 125 VDC Diesel Generator Auxiliary Power System with increased capacity, extended service life, and improved reliability. The additional weight of the batteries has been qualified for the rack and floor loadings including seismic considerations. Charger sizing and Diesel Building Ventilation has been evaluated to be acceptable. Cable sizing and circuit breaker interrupting capacity has been evaluated to be acceptable.

Proper design considerations have been made to assure that no new failure modes are created. The resulting design is more reliable without placing any unacceptable demands on supporting systems. No unreviewed safety questions are introduced by the modification. No Technical Specification changes are required. Changes are required to UFSAR Section 8.3.2.1.2.2 and 9.4.4.2.

4 Type: Nuclear Station Modification **Unit:** 1

Title: Nuclear Station Modification CN-11389/00, Backup Cooling for the Chemical and Volume Control System Centrifugal Charging Pump 1A Motor Coolers and Pump Oil Coolers

Description: Nuclear Station Modification CN-11389/00, provides a backup cooling source to support components of the Chemical and Volume Control System Centrifugal Charging Pump 1A. The backup cooling water will come from the non nuclear safety related Drinking Water System. It will supply the Centrifugal Charging Pump 1A Motor Cooler, Speed Reducer Cooler, and Pump Bearing Oil Cooler. Normal cooling to the Centrifugal Charging Pump 1A is supplied by the Component Cooling System Essential Supply Header. The Backup Supply from the Drinking Water System will be tied into the existing Component Cooling Supply Header using a backflow preventer. On the Component Cooling System return header, drain lines will be installed to route the Drinking Water System flow to the Residual Heat Removal /Containment Spray Pump Room Sump. In addition, an alternate discharge connection will be provided to allow diversion of the return flow to another location (for example, the Auxiliary Building Groundwater Drainage Sump C via a firehose) if desired and approved by Radwaste Chemistry.

Evaluation: The purpose of this modification is to reduce the overall core melt frequency as calculated by the Catawba Probabilistic Risk Assessment. Loss of Component Cooling and Nuclear Service Water are significant contributors to Reactor Coolant Pump Seal Loss of Coolant Accidents (LOCA) and the subsequent fuel damage. Loss of Nuclear Service Water will eventually lead to loss of Component Cooling and hence a loss of cooling to the Centrifugal Charging Pumps and to the Reactor Coolant Pumps thermal barrier heat exchangers.

The Drinking Water System tie-ins to the Component Cooling System will maintain the required seismic integrity of Component Cooling by using nuclear safety related, Duke Class C manual isolation valves. This modification does not add any new water supplies, but utilizes the existing Drinking Water System source in the Auxiliary Building. Therefore, no new flood sources are introduced. No electrical modifications are being made therefore there is no electrical impact on design basis events.

This modification does not involve an unreviewed safety question. No Technical Specification changes are required. UFSAR Section 9.3.4.2.3.1 will be revised to describe the backup cooling of the 1A Centrifugal Charging Pump. New UFSAR Sections 9.2.2.3.7 (Component Cooling) and 11.2.2.2.4.3.1 (Liquid Radwaste) are being created to describe their role in providing backup cooling to the 1A Component Cooling Pump. UFSAR Figure 11-2 (Liquid Radwaste System flow drawing) is being revised to show the two inch Drinking Water System line coming into the Residual Heat Removal /Containment Spray Pump room sump.

5 Type: Nuclear Station Modification **Unit:** 1

Title: Nuclear Station Modification CN-11391/00, Modify Auxiliary Feedwater System Control Valves to prevent Overfill during Steam Generator Tube Rupture

Description: Nuclear Station Modification CN-11391/00 modifies Auxiliary Feedwater System Control Valves to prevent overfill during Steam Generator Tube Rupture. This modification will provide accumulator air tanks and instrumentation tubing for the Auxiliary Feedwater System flow control valves to provide for the operation of these valves to isolate Auxiliary Feedwater to any Steam Generator which might have experienced a tube rupture following a Loss of Offsite Power and Safety Injection. Train A of Motor Driven Auxiliary Feedwater supplies water to Steam Generators A and B. Train B of Motor Driven Auxiliary Feedwater supplies water to Steam Generators C and D. The Turbine Driven Auxiliary Feedwater Pump supplies water to all four Steam Generators. Since there is one valve in each line, there are a total of eight flow control valves. One air tank will be provided per valve. These tanks will "ride' the Instrument Air System and will be isolated from the effects of failures in the Instrument Air System by two check valves in series. All tanks, tubing and check valves are nuclear safety related. The time period for which this air will be available is 60 minutes. These components will provide protection for Steam Generator Overfill Prevention for Design Basis Steam Generator Tube Rupture with certain other failures.

Evaluation: This station modification modifies the Auxiliary Feedwater System which is an accident mitigation system. No changes are being made to the actuation circuitry for the Auxiliary Feedwater System. There are no UFSAR Chapter 15 events which assume that the Auxiliary Feedwater System contributes as an accident initiator. Therefore no new failure modes are created and no accidents of a different type than those evaluated in the UFSAR are created. This modification serves as a partial solution to the problem of the inability to terminate Auxiliary Feedwater System flow to prevent Steam Generator overfill following a Steam Generator Tube Rupture. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. UFSAR Section 10.4.9 and UFSAR Table 9-17 will need to be revised.

229 **Type:** Nuclear Station Modification **Unit:** 2

Title: Nuclear Station Modification CN-20571/00, Replace Power Supplies and Circuit Breakers in Radiation Monitor Cabinets

Description: Nuclear Station Modification CN-20571/00 will replace the DC Power Supplies for Radiation Monitor Cabinets with new DC Power Supplies. Also all 2.5 Amp circuit breakers will be replaced with 5.0 Amp breakers. Currently, each radiation monitor channel uses an individual 24 Volt Linear DC power supply which is inefficient. This causes an increased temperature in the cabinets which could adversely affect the operation of the monitors.

Evaluation: There are no unreviewed safety questions associated with this modification. The radiation monitors have no effect on accidents analyzed in the UFSAR. No new accident scenarios are created. No seismic concerns are created by the changeout of these power supplies. This modification will not increase the load on the power supplies. There are no Appendix R concerns because no cables are being routed. No Technical Specification changes are required. No UFSAR changes are required.

40 **Type:** Nuclear Station Modification **Unit:** 2

Title: Nuclear Station Modification CN-21379 Replace 6.9 KV Switchgear Tie Breakers with Faster Vacuum Breakers

Description: Nuclear Station Modification CN-21379 Revision 0 will replace the current 6900 Volt tie breakers with faster vacuum breakers, because the current fast transfer schemes have the potential to subject the switchgear loads (motors) to excessive restart torque. The modification will also install switches to provide the function of defeating the automatic fast transfer when the units are off line.

The Unit Main Power System is part of the Onsite Power System which directly interfaces with the Offsite Power System. The balance of the Onsite Power System consists of the Diesel Generators, batteries, controls and auxiliary power system. The Unit Main Power System includes the main generator, isolated phase busses, Generator Power Circuit Breakers and associated motor operated disconnects, main step-up transformers, four unit auxiliary (20.9/6.9 KV) transformers and one auxiliary (20.9/13.8 KV) transformer. The Unit Main Power System starts with the main generator which feeds two trains (A and B) of transformers, breakers and conductors. The main generator feeds: two "20.9 KV to 230 KV half sized" unit step up transformers, and four "20.9 KV to 6.9 KV half sized" unit step down auxiliary transformers, and one auxiliary "20.9 KV to 13.8 KV step down" transformer through the isolated phase busses and two Generator Power Circuit Breakers. The Offsite Power System consists of the entire station switchyard including Power Circuit Breakers, associated Motor Operated Disconnects, conductors, and protective relaying.

Power is fed from the Onsite Power System to the 6900 Volt Normal Auxiliary Power System. The 6900 Volt Normal Auxiliary Power System includes four switchgear assemblies of a split bus design including a short leg split from a long leg on each bus. The long and short legs of a respective switchgear are normally powered from opposite trains of the Onsite Power System. These switchgear assemblies are named 2TA, 2TB, 2TC, and 2TD. Normal power for these switchgear is supplied from 20.9/6.9 KV transformers 2T1A, 2T2A, 2T2B, and 2T1B. In order of short leg then long leg, the power sources on 2TA, 2TB, 2TC, 2TD are trains AB, BA, AB, BA.

The 6900 Volt tie breakers provide the capability to tie together the short and long leg of a particular switchgear assembly such that both legs are powered by the same train of the Onsite Power System. The 6900 Volt Normal Auxiliary Power System feeds the 4160 Volt Essential Auxiliary Power System. Normal alignment has train A 20.9/6.9 KV transformer 2T2A feeding switchgear 2TA (short leg) which feeds Essential bus 2ETA through 6.9/4.16 KV transformer 2ATC; and train B 20.9/6.9 KV transformer 2T1B feeding switchgear 2TD (short leg) which feeds Essential Bus 2ETB through 6.9/4.16 KV transformer 2ATD.

It is through these 6900 Volt tie breakers that the 4160 Volt Essential busses can be supplied from the opposite 6900 Volt train. A "Zone Lockout" is protective action provided in the switchyard that isolates one side of the Offsite Power System including the generator from a fault by opening the switchyard and associated Generator Power Circuit Breakers. Following this isolation, the tie breakers can power the de-energized

6900 Volt switchgear from the opposite train of Unit Main Power System with the resulting alignment having one 6900 Volt switchgear completely supplied from one auxiliary transformer. This automatic transfer will also occur for any reason an under voltage condition exists on a particular short or long leg provided the under voltage is not due to a fault on the bus.

The subject tie-breakers are non-safety related devices. Essential Power is provided to the safety related 4160 volt Essential Auxiliary Power System via the Class 1E Diesel Generators. Therefore, the normal power source to the station auxiliaries is mainly important to the extent that plant transients are not created through interruption of power. Interruption of power through the normal feed paths discussed above (2ATC and 2ATD), would create a blackout condition such that the load sequencer would start and load the Diesel Generator onto the affected Essential Bus. Additionally, interruption of power would terminate powered operation of the reactor coolant pumps and they would begin a coastdown due to the rotating inertia of the flywheel. Loss of forced flow in one loop will cause a Reactor Trip from an initial power level exceeding 48 % (P8). Below 48 % but greater than 10 % (P10), two loops must lose forced flow to result in a Reactor Trip. Per Tech Spec Table 3.3.1-1 and UFSAR Table 7-1, a 2 out of 4 undervoltage condition on reactor coolant pump motor voltage, will cause a Reactor Trip at less than or equal to 77 % normal voltage per UFSAR Section 7.2.1.1.2. This protection is provided by the Reactor Coolant Pump Monitor System. Unsuccessful transfer of power will result in a loss of flow. A slow transfer of power will result in a Reactor Trip. A Reactor Trip causes a Turbine Trip from any power level if the Turbine is not tripped. Thus, the 6900 Volt Normal Auxiliary Power System is identified as an accident initiator. The Unit Main Power System is also an accident initiator since its transformers feed the 6900 Volt switchgear.

Additionally, through auto-starting the Diesel Generators, challenges to accident mitigation equipment are increased if normal power equipment is made less reliable. Thus, the tie breakers are evaluated to be equipment important to safety as they can initiate a transient (Reactor Trip/ Turbine Trip) and cause challenges to accident mitigation equipment (Diesel Generators) if they fail to operate as designed when called upon. In addition to the above equipment other equipment such as Hotwell Pumps, Condensate Booster Pumps, and Condenser Circulating Water Pumps are powered from the 6900 Volt switchgear.

Evaluation: There are no unreviewed safety questions associated with this modification. All scenarios involving possible motor restart torque potential are improved by this modification due to the faster transfer time provided by the replacement breakers. The effects of this modification on breaker coordination has been evaluated. It was determined that the effects, if any, would be positive. No Technical Specification changes are required. A change is required for UFSAR Section 8.3.1.1.1.3.

248 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 addition of new actuators on Valves 2NS-38B and 2NV-252A to eliminate pressure locking concerns

Description: Nuclear Station 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 the Catawba Nuclear Station response to NRC Generic Letter 95-07.

Valves 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 1/2" vent path to the hot leg side of the valve. Removing Containment Valve Injection Water System connection was discussed with the NRC and its justification is documented in calculation CNC-1223.12.000062, Justification for Removal of the Containment Valve Injection Water System Supply from Valves N1121A and N1152B. The calculation shows that, even without the Containment Valve Injection Water System connection, 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. 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 would 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, 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 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 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 were 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 effects of the heavier actuator on the existing valve were evaluated by Westinghouse and the evaluation is documented in CNM 1205.19-0103.001, 0913-235 Seismic and Weak Link Analysis of Westinghouse Eight Inch 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 necessary is the replacement of the overload heaters located in motor control center 2EMXA.

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, are maintained.

Evaluation: Modification CN-21386/00 does not involve an unreviewed safety question. No Technical Specification changes are required. UFSAR 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 Unit 1 modification (CN-11385) is implemented.

168 **Type:** Nuclear Station Modification **Unit:** 2

Title: Nuclear Station Modification CN-21389/00, Backup Cooling for the Chemical and Volume Control System Centrifugal Charging Pump 1A Motor Coolers and Pump Oil Coolers

Description: Nuclear Station Modification CN-21389/00, provides a backup cooling source to support components of the Chemical and Volume Control System Centrifugal Charging Pump 2A. The backup cooling water will come from the non nuclear safety related Drinking Water System. It will supply the Centrifugal Charging Pump 2A Motor Cooler, Speed Reducer Oil Cooler, and Pump Bearing Oil Cooler. Normal cooling to the Centrifugal Charging Pump 2A is supplied by the Component Cooling System Essential Supply Header. The Backup Supply from the Drinking Water System will be tied into the existing Component Cooling Supply Header using a backflow preventer. On the Component Cooling System return header, drain lines will be installed to route the Drinking Water System flow to the Residual Heat Removal /Containment Spray Pump Room Sump. In addition, an alternate discharge connection will be provided to allow diversion of the return flow to another location (for example, the Auxiliary Building Groundwater Drainage Sump C via a firehose) if desired and approved by Radwaste Chemistry.

Evaluation: The purpose of this modification is to reduce the overall core melt frequency as calculated by the Catawba Probabilistic Risk Assessment. Loss of Component Cooling and Nuclear Service Water are significant contributors to Reactor Coolant Pump Seal Loss of Coolant Accidents (LOCA) and the subsequent fuel damage. Loss of Nuclear Service Water will eventually lead to loss of Component Cooling and hence a loss of cooling to the Centrifugal Charging Pumps and to the Reactor Coolant Pumps thermal barrier heat exchangers.

The Drinking Water System tie-ins to the Component Cooling System will maintain the required seismic integrity of Component Cooling by using nuclear safety related, Duke Class C manual isolation valves. This modification does not add any new water supplies, but utilizes the existing Drinking Water System source in the Auxiliary Building. Therefore, no new flood sources are introduced. No electrical modifications are being made therefore there is no electrical impact on design basis events.

This modification does not involve an unreviewed safety question. No Technical Specification changes are required. UFSAR Section 9.3.4.2.3.1 will be revised to describe the backup cooling of the 2A Centrifugal Charging Pump. New UFSAR Sections 9.2.2.3.7 (Component Cooling) and 11.2.2.2.4.3.1 (Liquid Radwaste) are being created to describe their role in providing backup cooling to the 2A Centrifugal Charging Pump. UFSAR Figure 11-2 (Liquid Radwaste System flow drawing) is being revised to show the two inch Drinking Water System line coming into the Residual Heat Removal /Containment Spray Pump Room sump.

191 **Type:** Nuclear Station Modification **Unit:** 2

Title: Nuclear Station Modification CN-21391/00, Modify Auxiliary Feedwater System Control Valves to prevent Steam Generator Overfill during Staem Generator Tube Rupture

Description: Nuclear Station Modification CN-21391/00 will provide accumulator air tanks and instrumentation tubing for the Auxiliary Feedwater System flow control valves. This will provide for the operation of these valves to isolate Auxiliary Feedwater flow to any Steam Generator that experiences a tube rupture following a Loss of Offsite Power and Safety Injection. The following eight flow control valves are involved in the modification 2CA36, 2CA40, 2CA44, 2CA48, 2CA52, 2CA56, 2CA60 and 2CA64. The Train A Auxiliary Feedwater Motor Driven Pump supplies auxiliary feedwater flow to Steam Generators A and B. The Train B Auxiliary Feedwater Motor Driven Pump supplies auxiliary feedwater flow to Steam Generators C and D. The Turbine Driven Auxiliary Feedwater Pump supplies auxiliary feedwater flow to all four Steam Generators. One air tank will be provided per valve. The air tanks will "ride" the instrument air system and will be isolated from the effects of Instrument Air system failures by two check valves in series. The air tanks will provide air for sixty minutes.
Steam Generator overfill has been identified as an area of concern for the two units at Catawba. An administrative limit has been placed on Dose Equivalent Iodine (DEI) as part of an Operable but Degraded (OBD) evaluation associated with this problem. This limit is required until such time that a combination of plant modifications and other changes, possibly including a license amendment, can be installed/obtained that will nullify the effects of certain failures, thereby making the temporary DEI limit unnecessary.

It has been determined that the Auxiliary Feedwater System should be capable of avoiding Steam Generator overfill conditions following a design basis Steam Generator Tube Rupture (SGTR) which involves a Loss of Offsite Power (LOOP) and a Safety Injection (SI) signal and certain other failures. The design of the Instrument Air System is such that the Diesel Generator power available from the blackout switchgear to the Instrument Air System compressors would be unavailable following a LOOP/SI scenario.

Interlocks exist which preclude the simultaneous energization of the blackout switchgear from both the essential and blackout auxiliary power system. Following an SI signal which isolates the essential switchgear from the blackout switchgear per UFSAR 8.3.1.1.2.1, along with the unavailability of normal power due to the LOOP, the blackout switchgear would be deenergized. Thus, instrument air would not be available, and the air operated Auxiliary Feedwater System control valves will eventually fail open.

Subsequent failure of an Emergency Diesel Generator to start or run appears to include loss of power to the following:

1. An Auxiliary Feedwater System Motor Driven Pump
2. Isolation valves in line from the failed Auxiliary Feedwater System Motor Driven Pump to its two S/G's.
3. Isolation valves in line from the Auxiliary Feedwater System Turbine Driven Pump to the opposite two S/G's.
4. One train of the Emergency Core Cooling System (ECCS) and the remaining ESF equipment (the "minimum safeguards" scenario).

The 125 VDC Vital Intrumentation and Control System batteries in the affected Class 1E train would supply power to low voltage (125 VDC/120VAC) loads for a minimum of two hours. The Class 1E low power loads include the remote controls for the S/G Power Operated Relief Valves (PORV's). DC loads supplied directly from distribution centers EDE/EDF would supply power from the Vital Batteries for a period of two hours per UFSAR 8.3.2.1.2.1.2.

This modification fixes the overfill problem for the above described scenario including the failure of a diesel generator, which is limiting for all failures except an EDE failure (for which a license amendment may be necessary). The temporary isolation of Auxiliary Feedwater using the backup air to the flow control valves will allow an operator to locally manually isolate Auxiliary Feedwater to the affected (ruptured) S/G.

If other failures occur (instead of the D/G failure) such as a specific Auxiliary Feedwater isolation valve (EMO), the flow of auxiliary feedwater would also not be controllable from the Control Room after Instrument Air becomes unavailable as described in the design basis event sequence above. The response to these failures would be the same: temporarily isolate the affected (ruptured) S/G via the flow control valves and dispatch an operator for local manual isolation via the EMOs .

Evaluation: The addition of the air tanks will provide the capability to close the flow control valves in those Auxiliary Feedwater lines which would otherwise be unisolable from the Control Room since the associated EMO would be de-energized due to Diesel Generator failure. The ability to close the flow control valve presumes the solenoid valves are energized, which they would be on the Auxiliary Feedwater train that has experienced the Diesel Generator failure, due to the vital batteries. The emergency operating procedures currently provide for dispatching personnel to the affected valves to permanently secure the isolation. The time period provided by the vital batteries and the new air tanks will be sufficient to ensure re-opening of the flow control valve does not occur in an unanticipated manner.

The eight (8) new air tanks and the associated connecting tubing and check valves are not nuclear safety related. Two check valves per air tank are installed to ensure that adequate air pressure is maintained for the flow control valves should the Instrument Air System be depressurized. No undesirable seismic interactions are created due to the addition of the air tanks as they seismically mounted even though they are non-QA. Also, no pipe rupture or interaction concerns are created with the routing of the tubing from the new air tanks to the individual flow control valves . No Appendix R concerns have been created since there is no cable routing or changes to any electrical power supplies. The air supply is designed for 60 minutes. The solenoid valves associated with the train that has lost the availability of a D/G are powered by the 125 VDC Vital Power which is designed for two hours per UFSAR 8.3.2.1.2.1.2. Thus, the isolation can occur and remain in effect until permanent isolation can be accomplished via existing Emergency Procedures.

The safety related function of the Auxiliary Feedwater System flow control valves is to fail open on loss of air or power to the actuators or open on receipt of an Auxiliary Feedwater System auto start signal. A safety grade solenoid in the valve's air line vents the valve operator which opens the valve. This capability is not altered in any way by this modification.

This modification modifies an accident mitigation system. No changes are being made to the Auxiliary Feedwater actuation circuitry. There are no UFSAR Chapter 15 events which assume that the Auxiliary Feedwater system acts as an accident initiator. No new failure modes or accident scenarios are created by this modification. The tanks are seismically mounted.

There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. Changes are required for UFSAR Section 10.4.9 and UFSAR Table 9-17.

249 **Type:** Nuclear Station Modification **Unit:** 2

Title: Nuclear Station Modification CN-21393/00, Main Feedwater Containment Isolation Valve Actuator Modification.

Description: Nuclear Station Modification CN-21393/00 will resolve problems associated with the Unit 2 Feedwater System Isolation valves (2CF-33, 2CF-42, 2CF-51, and 2CF-60). These valves are 18" Borg-Warner 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 Operator Aid Computer (OAC) alarms. Finally, the filters and orifices on the hydraulic solenoid valves will be replaced with transfer tubes to improve performance.

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 are 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 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 evaluation was performed to justify pneumatic solenoid valve removal. 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). In summary, no new failure modes are created. A potential failure has been 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 as nuclear safety related equipment.

Evaluation: There are no unreviewed safety questions associated with modification CN-21393/00. No Technical Specification changes are required. No changes to the UFSAR are required. The relevant accidents for the MFIVs are those resulting in the applicable actuation signals: Safety Injection, Steam Generator hi- hi level, Reactor Trip with lo-Tavg, and Doghouse Water level hi-hi - all of which 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. 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 to the Nitrogen Tanks.

2 **Type:** Nuclear Station Modification **Unit:** 0

Title: Nuclear Station Modification CN-50458/00, Install Conventional Waste Water Treatment System Automatic pH Release Valve and Upgrade the associated pH Instrumentation

Description: Nuclear Station Modification CN-50458/00 will install an automatic pH Release Valve on Conventional Waste Water Treatment System and Upgrade the associated pH Instrumentation.

The Conventional Waste Water Treatment System is designed to condition non-radioactive non-sanitary waste for discharge to Lake Wylie in compliance with Federal (EPA) and/or State of South Carolina discharge water quality guidelines. It accomplishes these functions utilizing three major stages of processing: initial holdup pond, settling ponds A and B, and the final holdup pond.

During releases to the lake, periodic sampling is performed to assure the pH of effluent is within limits specified by the NPDES permit. There is an alarm in the Water Chemistry Building for "hi" and "lo" pH. Upon alarm receipt, the Chemistry Technician takes a confirmatory sample of the basin prior to terminating the release. Chemistry personnel availability is such that Water Treatment and Secondary areas get first priority attention particularly on weekends and night shift. Therefore, an automatic pH monitoring and release termination system will be installed in order to alleviate the concern of a potential NPDES violation due to low staffing levels.

There is a normally closed electric motor operated valve (1WC-225) installed in the discharge piping to the lake. This valve is located in the Final Holdup Pond Inlet and System Outlet Valve Pit and downstream of the discharge line pH instrumentation. During releases to the lake when the valve is opened, this valve will automatically close on emergency high or emergency low pH in the discharge stream. This valve will have Open/Closed indication in the Water Chemistry Building with an Open/Closed/Auto control switch.

Additionally, UFSAR Figure 2-4 and associated text in UFSAR section 2.1.1.3 will be modified to include the Conventional Waste Water Treatment System discharge point which is not currently shown on UFSAR Figure 2-4. This change is not directly attributable to this modification which is concerned with the pH of effluents discharged to the lake. However, this UFSAR figure change is relevant to this modification which is concerning releases to the lake from the Conventional Waste Water Treatment System and this is a convenient time to correct this oversight in the UFSAR.

Additionally, per Variation Notice CN-50458D, UFSAR Figure 16.11-1 (Selected Licensee Commitment) will be revised to show the Conventional Waste Water Treatment System discharge location which is not currently shown.

Evaluation: This modification does not introduce any Unreviewed Safety Questions. The Conventional Waste Water Treatment System is not an accident initiator and serves no role in accident mitigation. No Technical Specification changes are required. UFSAR changes are required to Section 2.1.1.3 and UFSAR Figure 2-4 and UFSAR Figure 16.11-1 (Selected Licensee Commitment).

1 **Type:** Procedure **Unit:** 0

Title: Changes to procedures PT/1(2)/A/4450/001D, PT/0/A/4450/001B, PT/0/A/4450/001C, PT/0/A/4450/004B, PT/1(2)/A/4450/001A, PT1(2)/A/4450/001E, and PT/1(2)/A/4450/007A

Description: Procedures PT/1(2)/A/4450/001D, PT/0/A/4450/001B, PT/0/A/4450/001C, PT/0/A/4450/004B, PT/1(2)/A/4450/001A, PT1(2)/A/4450/001E, and PT/1(2)/A/4450/007A were changed to incorporate the DOP Photometer TDA-2GN Operating Instructions. These procedures provide requirements to perform filter train performance tests to the following ventilation filter systems: Auxiliary Building Filtered Exhaust System, Control Room Area Ventilation System, Annulus Ventilation System, Fuel Handling Ventilation Exhaust System, Technical Support Center Ventilation System, Containment Purge Exhaust System, Containment Air Release and Addition System.

Evaluation: These procedures furnish instructions on how to perform filter testing per ANSI N510-1980. These changes add another model of a piece of test equipment. The two models perform identical tasks which is to sample air or gas and react to any particulate drawn through the scattering chamber. The filter units associated with these procedures are not accident initiators. This procedure changes does not change the testing method or acceptance criteria. There is no unreviewed safety question associated with these procedure changes. No Technical Specification changes are required. No UFSAR changes are required.

64 **Type:** Procedure **Unit:** 0

Title: Procedure AP/1(2)/A/5500/06, "Loss of Steam Generator Feedwater" Revision 25 (Unit 1) and Revision 20 (Unit 2)

Description: Procedure AP/1(2)/A/5500/06 "Loss of Steam Generator Feedwater" Revision 25 (Unit 1) and Revision 20 (Unit 2) makes changes to "Case II, Loss of Normal Auxiliary Feedwater Supply" to make hotwell suction methods consistent with assumptions in calculation CNC-1223.42-00-0022 and an operability evaluation associated with Problem Investigation Report 0-C90-0079. This will involve five changes:

1) Remove the 8.2 psig Auxiliary Feedwater Pump suction pressure manual trip criteria and add note to state that "Auxiliary Feedwater Pump suction pressure will decrease from approximately 11 psig at normal hotwell level and 600 gpm to approximately 8 psig at 600 gpm at .5 ft hotwell level". The reason for this change is the 8.2 psig auxiliary feedwater suction pressure Auxiliary Feedwater Pump manual trip criteria could result in premature pump trip if the intent is to utilize the maximum hotwell inventory. A 1 psig error in Auxiliary Feedwater Pump suction pressure is equivalent to 2.3 feet of hotwell level.

2) Add notes to ensure local indications of Hotwell level, Upper Surge Tank (UST) level and vacuum are used when offsite power is not available since these indications do not receive long term battery backed power.

3) Require that all four S/G levels have been restored to normal levels prior to initiating swap to the hotwell. This assures that the heat sink Critical Safety Function is met prior to securing Auxiliary Feedwater Pumps.

4) Add a restriction to limit total Auxiliary Feedwater System flow to less than 600 gpm consistent with the flow assumed in calculation CNC-1223.42-00-0022, Operability Evaluation for PIR 0-C90-0079, while taking suction from the Hotwell.

5) Require securing operating Auxiliary Feedwater Pumps at 10% UST level and closing valve CA4 prior to restarting the Auxiliary Feedwater Pumps. This ensures that air cannot be admitted to the Auxiliary Feedwater System suction line, in the event that Auxiliary Feedwater System flowrate exceeds 700 gpm while the system is aligned to the hotwell.

Evaluation: The purpose of Case II of procedure AP/1(2)/A/5500/0 is to align condensate sources to the Auxiliary Feedwater Pumps and align the Nuclear Service Water System assured source when the condensate sources are depleted. These changes ensure the safety related Auxiliary Feedwater Pumps are protected from damage while allowing the non safety related condensate storage system (CSS) to supply the steam generators (S/G's) during a LOOP event per Improved Technical Specification (ITS) 3.7.6.

The Auxiliary Feedwater System is a nuclear safety related system which functions to remove heat from the Steam Generators and to allow cooldown of the Reactor Coolant System to the point where Residual Heat Removal System can be placed in service. The system is required to be OPERABLE in Modes 1, 2, and 3 and in Mode 4 while steam generators are utilized for decay heat removal, per Technical Specification 3.7.5. This function may be required for normal operation and for UFSAR Chapter 15 Accident Mitigation. The Auxiliary Feedwater System by design has five suction sources which may be aligned. These are:

1. Auxiliary Feedwater Condensate Storage Tank (Non-Safety Related)
2. Upper Surge Tank (Non-Safety Related)

3. Hotwell (Non-Safety Related)
4. Nuclear Service Water System (Safety Related Assured Source)
5. Condenser Cooling Water (Non-Safety Related)

The Auxiliary Feedwater System is designed so that the normal sources are the non-nuclear safety related condensate grade sources from the CACST or the UST. Currently, the Auxiliary Feedwater Storage Tank (CACST) is normally isolated. Therefore, the Upper Surge Tank is currently the normal source for the Auxiliary Feedwater System. Under accident conditions if the non-safety sources are not available the suction will automatically swap to the assured source, the Nuclear Service Water System. Non-safety suction source availability is determined by having adequate pressure from these sources on safety related pressure switches, with two out of three logic, in Auxiliary Feedwater Suction piping. Manual Operator actions are required to use the hotwell. These are defined in AP/1(2)/A/5500/06, Loss of S/G Feedwater. The required actions include, breaking condenser vacuum, defeating Auxiliary Feedwater System to Nuclear Service Water System Autoswap.

The Upper Surge Tank and the Hotwell are part of the Condensate System which functions as part of the Secondary Side to supply heated condensate quality water to the S/Gs for steam and power generation. These systems are not safety related and not required to serve an accident mitigation function. The hotwell functions as the collection point for all secondary side condensate and as such is the suction source for the Hotwell Pumps in addition to providing a suction source for the Auxiliary Feedwater System.

ITS 3.7.6, Condensate Storage System (CSS), requires CSS inventory of 225,000 gallons be maintained. The basis of the CSS inventory is to maintain the condensate grade inventory required to maintain hot standby for two hours followed by a five hour cooldown to Mode 4 following a LOOP event.

In 1990, PIR (Corrective Action Program) Serial Number 0-C90-0079 evaluated the use of the hotwell as a condensate quality source for the Auxiliary Feedwater System. This PIR was initiated as a result of an Auxiliary Feedwater Pump low suction pressure trip that resulted during a retest on check valve 1CA1 while the Auxiliary Feedwater System was taking suction from the Condenser Hotwell. The calculation determined that the non safety related Auxiliary Feedwater Pump low suction pressure trips must be defeated and main condenser vacuum must be broken in order to use the maximum inventory of the hotwell. The inventory of the hotwell is needed in order to meet the intent of the Condensate Storage System (CSS) Specification, original Technical Specification 3.7.1.5, which applied only to Unit 2. The CSS specification is now ITS 3.7.6 and applies to Unit 1 and Unit 2. As a result of PIR 0-C90-0079, Procedure AP/1(2)/A/5500/06 was revised in 1990 to allow defeat of the non safety related Auxiliary Feedwater Pump low suction pressure trip following depletion of the higher head condensate sources. A manual trip criteria of 3.5 psig suction pressure or .5 foot indicated hotwell level was established. The Auxiliary Feedwater pump suction pressure criteria was established due to limited indication of hotwell level in 1990. Since that time, a Hotwell level sight glass has been added that provides hotwell level indication in a LOOP event. The control room indication on chart recorder 1(2)CSCR5840 does not receive battery backed power. PIP 0-C91-0074 changed the Auxiliary Feedwater Pump suction pressure trip criteria setpoint from 3.5 psig to 8.2 psig since the basis for the 3.5 psig criteria was not documented. At the time PIP 0-C91-0074 was evaluated, the hotwell level sight-glass had not been installed.

The changes to Procedure AP/1(2)/A/5500/06 in this evaluation make operation of Auxiliary Feedwater while taking suction from the condenser hotwell consistent with the operability evaluation documented in calculation CNC-1223.42-00-0022, Operability Evaluation for PIR 0-C90-0079 and consistent with the ITS Condensate Storage System specification basis. The addition of the local hotwell level sight-glass 1(2)CMLG5450 provides a reliable hotwell level termination criteria.

Item 1

Deleting the 8.2 psig Auxiliary Feedwater pump suction pressure manual trip criteria does not involve an unreviewed safety question for the following reasons. Calculation CNC-1223.42-00-0022, Operability Evaluation for PIR 0-C90-0079, recommends that Auxiliary Feedwater pump trip criteria be based entirely on hotwell level. The hotwell level indication is acceptable, as a termination criteria, since the hotwell level decreases slowly during the event due to the physical characteristics of the Hotwell, thus allowing operators adequate time to respond to the pump trip criteria. The hotwell level and piping between the hotwell and Auxiliary Feedwater pump room is not safety related. However, failure of this piping is not assumed following an event that would cause an Auxiliary Feedwater auto start provided the piping did not fail prior to or during the event. This assumption is documented in calculation CNC 1223.42-00-0001, Nuclear Service Water Transfer Scheme Adequacy.

Item 2

Item 2 changes Procedure AP/1(2)/A/5500/06 to ensure that the operators are using reliable indications for hotwell level and verification that vacuum has been broken in the event offsite power is lost. These locations are accessible in the event of a loss of offsite power.

Item 3

Item 3 changes Procedure AP/1(2)/A/5500/06 to require that Steam Generator levels have been restored to normal levels prior to initiating swap to the hotwell. This assures that the Heat Sink Critical Safety Function is satisfied prior to tripping the Auxiliary Feedwater pumps. Tripping of the Auxiliary Feedwater Pumps while a verifiable level is in the UST is a required action for transferring Auxiliary Feedwater suction to the Hotwell. Tripping the pumps and closing valve CA4 prevents the possibility of air being introduced to the Auxiliary Feedwater suction line if Auxiliary Feedwater flows exceed 700 gpm (per a vendor supplied analysis). Requiring normal S/G levels prior to transferring suction to the Hotwell is acceptable, since the UST volume is sufficient to recover S/G level in a LOOP or Loss of Normal Feedwater accident. LOOP and Loss of Normal Feedwater accidents are the basis for CSS. If S/G levels are not above normal levels by the time transfer to the Hotwell is required the accident is beyond the basis for maintaining condensate sources and therefore transfer to the assured Nuclear Service Water System source is required.

Item 4

Item 4 adds a restriction to limit total Auxiliary Feedwater flow to less than 600 gpm consistent with flow assumed in calculation CNC-1223.42-00-0022, Operability Evaluation for PIR 0-C90-0079 while taking suction from the Hotwell. The CSS specification basis is to assure a condensate supply for Condition 1 and 2 Transients, specifically a LOOP, to prevent introduction of Nuclear Service Water to the S/G's. Per calculation CNC-1223.42-00-0022, Operability Evaluation for PIR 0-C90-0079, by the time transfer to the Hotwell occurs, the decay heat removal requirements and cooldown requirements will be less than 600 gpm. If the cooldown requirements are greater than 600 gpm the operators are instructed to swap suction to the safety related Nuclear Service Water System assured source.

Item 5

Require securing the operating Auxiliary Feedwater Pumps to 10% UST level and closing valve CA4 prior to restarting the Auxiliary Feedwater Pumps. By the time the UST is depleted at 10% UST level the S/G levels should have recovered to the minimum acceptable levels during Condition 1 and 2 Transients for which the CSS specification is based. Securing the Auxiliary Feedwater Pumps at 10% UST level assures that there is time to close valve CA4, Auxiliary Feedwater Suction from UST Isolation, prior to allowing the Auxiliary Feedwater suction line downstream of valve CA4 to fill with air. This prevents the possibility of air entering the Auxiliary Feedwater suction after the Auxiliary Feedwater suction is transitioned to the hotwell and the Auxiliary Feedwater Pump protective trips are defeated. As stated in item two, a prerequisite for transferring Auxiliary Feedwater suction to the Hotwell is that all four S/G levels are above the LO LO level setpoint. This assures that adequate heat sink is available prior to securing the Auxiliary Feedwater Pumps.

The purpose of AP/1(2)/A/5500/06 is to align condensate grade sources to the Auxiliary Feedwater suction and align these sources to the assured source when the condensate grade sources are depleted. AP/1(2)/A/5500/06 provides instruction to align Auxiliary Feedwater sources following accidents requiring Auxiliary Feedwater that are evaluated in the SAR. The condensate sources are not credited in the SAR. The SAR assumes the Auxiliary Feedwater suction is taken from the assured Nuclear Service Water System source. AP/1(2)/A/5500/06 and these changes to it would not increase the probability of an accident evaluated in the SAR since it only affects accident mitigation strategy and not accident initiation.

The changes to AP/1(2)/A/5500/06 decrease the probability of malfunction of equipment important to safety. As stated in the evaluation AP/1(2)/A/5500/06 aligns the suctions of Auxiliary Feedwater which is a system important to safety. The changes ensure the Auxiliary Feedwater System is operated under the assumptions originally evaluated in calculation CNC-1223.42-00-0022, Operability Evaluation for PIR 0-C90-0079. This assures that adequate Net positive suction Head (NPSH) is always available to the Auxiliary Feedwater Pumps to prevent damage while aligned to the low head Auxiliary Feedwater suction source. The deletion of the Auxiliary Feedwater Pump suction pressure manual trip criteria does not increase the probability of malfunction of the Auxiliary Feedwater Pumps. The hotwell level is adequate per the vendor analysis discussed above.

Procedure AP/1(2)/A/5500/06 is used following accident initiation during the recovery from loss of feedwater or any other accident requiring Auxilary Feedwater initiation. The change does not increase the radiological consequences since the changes insure the Auxiliary Feedwater Pumps are not damaged and therefore remain capable of cooling down the Unit to Mode 4.
No different type of accidents would be created by this change and no new failure modes are introduced. Therefore there are no unreviewed safety questions associated with this procedure change. No Technical Specification changes are required. No UFSAR revisions are required.

170 **Type:** Procedure **Unit:** 0

Title: Procedure HP/0/B/1001/030, Operation and Calibration for Zinc Sulfide (ZnS) Alpha Counting Systems, Revision 0

Description: Procedure HP/0/B/1001/030, Operation and Calibration for Zinc Sulfide (ZnS) Alpha Counting Systems, Revision 0 is a new procedure.

The purpose of the procedure is as follows:

1. To provide guidance for operating and calibrating the ZnS smear counter (automatic planchet counter (APC)).

2. To ensure that when power is lost to a ZnS smear counter instrument, it shall be re-energized properly without damage to the instrument.

3. To calculate results from the acquired data.

The ZnS Alpha Counting Systems are two systems. Each system has an alpha scintillation detector for counting alpha activity from smears and samples. The ZnS systems will replace the current Alpha Counting Systems which are defined in the USFAR Section 12.5.2.1.1. The ZnS Alpha Counting Systems will not require the use of P10 gas which is required for operation by the existing proportional Alpha Counting Systems. The ZnS Alpha Counting Systems are newer and enhanced systems which reflect the current state of the art technology for this type of equipment. The new systems will be more efficient to operate and calibrate. Both, the new ZnS Alpha Counting Systems and the old Alpha Counting Systems are not safety related and do not interface with any safety related systems.

Evaluation: The implementation of the Zinc Sulfide (ZnS) Alpha Counting Systems will not affect any plant systems, structures or components (SSC's) directly or indirectly. Therefore, no safety functions, design bases, or regulatory commitments will be affected for any SSC. There will be no affect on any accident analysis parameters and margins of safety.

The implementation of the Zinc Sulfide (ZnS) Alpha Counting Systems does not involve any of the following:

1. The addition or deletion of an automatic or manual feature of any SSC.
2. The conversion of an automatic feature to manual, or vice versa.
3. The introduction of an unwanted or previously unreviewed system interaction.
4. The alteration of the seismic or environmental qualification of any SSC.
5. The group classification of an SSC.
6. The change of an SSC which could affect core reactivity.
7. The activity on one unit that could affect another unit.

There is no Unreviewed Safety Question because the implementation of the ZnS Alpha Counting systems will have no impact on the ability to shut down the plant or maintain the plant in a shut down condition. These systems are not safety related and do not interface with any safety related systems.

No changes to the Technical Specifications are required. The USFAR Section 12.5.2.1.1 will be revised to reflect the ZnS Alpha Counting Systems.

104 **Type:** Procedure **Unit:** 0

Title: Procedure MP/0/A/7450/080, Revision 4, Troubleshooting and Corrective Maintenance of HVAC Dampers

Description: Procedure MP/0/A/7450/080, Revision 4 will provide guidance to allow manual volume dampers located upstream of the Fuel Pool Ventilation System filter units to be adjusted in small increments to adjust the air flow through the filter units. Each time the manual volume dampers are adjusted the air flow for the train will be tested to ensure that the train air flow meets Technical Specification Surveillance 3.7.13.3 flow requirements. The sequence of incremental adjustments will continue until the air flow meets the acceptance criteria of Fuel Pool Ventilation System filter train performance test procedures PT/1(2)/A/4450/001E or PT/1(2)/A/4450/009C.

Evaluation: There are no unreviewed safety questions associated with this procedure revision. This procedure revision does not increase the probabilty or consequences of an accident. The flows are maintained within the acceptance criteria of Technical Specification 3.7.13. The postulated accidents in the Spent Fuel Storage Building consist of a fuel handling accident and dropping of a weir gate onto a fuel assembly in the spent fuel pool. The proposed procedure change does not affect these accidents. The Spent Fuel Pool Ventilation System is not an accident initiator. No Technical Specification changes are required. No UFSAR changes are required.

99 **Type:** Procedure **Unit:** 0

Title: Procedure MP/0/A/7450/086 Revision 0 "Control Room Ventilation Ductwork and Air Handling Unit Access,"

Description: Procedure MP/0/A/7450/086 Revision 0 "Control Room Ventilation Ductwork and Air Handling Unit (AHU) Access," is a new maintenance procedure that was developed to provide guidance during 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/7450/086 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.

233 **Type:** Procedure **Unit:** 0

Title: Procedure MP/0/A/7450/087 Revision 0

Description: Procedure MP/0/A/7450/087 is a new procedure which will provide guidance for lubrication and adjustment of Auxiliary Building Ventilation System vortex dampers 1ABF D-6, 1ABF D-13, 2ABF D-6 and 2ABF D-13. These vortex dampers are located at the inlet of the Auxiliary Building Ventilation System filtered exhaust fans. These dampers are open during normal plant operation and go to a throttled position during a design basis accident. This procedure is needed for the adjustment of the dampers during flow balancing of the system. Performance testing (using procedures PT/0/A/4450/01C and PT/0/A/4450/04A) will be completed following adjustment of vortex dampers to ensure the Auxiliary Building Ventilation System continues to meet the requirements of Technical Specifications.

Evaluation: There are no unreviewed safety questions associated with this new procedure. The Auxiliary Building Ventilation System is not an accident initiator and this procedure will not prevent the system from maintaining the normal alignment ventilation exhaust air from the Auxiliary Building. Following any use of the procedure the Auxiliary Building Ventilation System will be tested to ensure that it meets Technical Specification requirements. The procedure will not increase the probability of accidents analyzed in the UFSAR and no new accidents are created. No Technical Specification changes are required. No UFSAR changes are required.

37 Type: Procedure **Unit:** 0

Title: Procedure MP/0/A/7650/151 Revision 0

Description: Procedure MP/O/A/7650/151, Rev 0, "Reactor Coolant (NC) Pump Motor Coolers' Corrective Maintenance" provides instructions for disassembly, inspection and restoration of the Reactor Coolant Pump Motor Coolers to support refurbishment plans of spare Reactor Coolant Pump Motors and any troubleshooting inspections for emergent needs.

Evaluation: UFSAR Section 9.2.2.3 requires all essential components of the Component Cooling System to be seismically designed and tested to meet ASME III, Class 3, except specified isolation valves. The Reactor Coolant Pump Motor Upper Bearing Oil Coolers are, however, on the non-essential header of the Component Cooling System as described in UFSAR 9.2.2.2 Component Cooling System cooling water is supplied to a number of components that are not essential to safe plant shutdown following a loss of coolant accident (LOCA) or steam break accident. Non-essential equipment can be supplied cooling water from either train of Component Cooling and, also, returned to either train of Component Cooling. But in the event of an engineered safeguards actuation signal during a LOCA or Steam Break Accident, motor operated isolation valves are actuated closed to provide separation of nonessential equipment from essential equipment.

In the event of a component failure that would result in out-leakage of Component Cooling System water on any of the components of the non-essential header, the Component Cooling system surge tanks have instrumentation which automatically separates the essential trains of Component Cooling components and isolates the non-essential headers upon low-low level in either surge tank. With the above Component Cooling surge tank inventory protection combined with train separation upon engineered safety signals, redundancy is provided for a single passive failure in the form of out-leakage anywhere in the Component Cooling system.

Therefore, failure of a Reactor Coolant Pump Motor Upper Bearing Oil Cooler manifesting in a loss of Component Cooling System inventory would result in low-low surge tank level and isolation of the failed cooler by isolation of its non-essential header.

As to further safety design of the Motor Lube Oil Coolers, they comply with the Standard Review Plan criteria, Section 9.5.1. The requirement is for the Reactor Coolant Pumps to have an oil collection system, which if it did fail, would not lead to fire during normal or design basis accident conditions and reasonable assurance of withstanding a safe shutdown earthquake.

The Reactor Coolant Pump Motor Air Coolers are Non-QA Condition, but the maintenance instructions were included in this subject QA related Maintenance Procedure to assure uniform type requirements for both coolers applicable for support of a common motor. The motor air coolers are mounted on the sides of the motor and are a copper alloy tube, plate-fin, construction with cooling supplied by the Containment Chilled Water System.

Based on the above safety review, performing maintenance on Reactor Coolant Pump Motor Coolers to restore them to design requirements specified in subject Maintenance Procedure does not degrade the reliability and/or safety function of the Component

Cooling System.

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

23 **Type:** Procedure **Unit:** 0

Title: Procedure OP/1(2)/A/6100/01, Controlling Procedure for Unit Startup

Description: Procedure OP/1(2)/A/6100/01 "Controlling Procedure for Unit Startup", other supporting systems and surveillance procedures, and UFSAR Sections 3.9.1.1 "Design Transients" and 5.3.3.1, "Reactor Vessel Design" were revised to allow higher administrative limits for Reactor Coolant System and Pressurizer Heatup and Cooldown to be implemented.

This change to the Operating Procedures and corresponding UFSAR sections will implement administrative Heatup and Cooldown limits that are approximately 80% of the licensing basis values.

In the Operating Procedures, administrative heatup and cooldown limits for the Reactor Coolant System and the Pressurizer are generally around 50% of the Technical Specification Limits. Recently, changes for outage optimization were made to one of these limits, (Reactor Coolant System Cooldown) to move the Administrative limit up to 75 degrees F./hr. cooldown, or 75% of the Technical Specification Limit. A review of how these "limits" are understood revealed inconsistencies. Some operators attempt to operate near the limit, while others operate at about half of these values. This caused difficulty in planning an outage. Through the use of newer technology, the operators are now able to more precisely follow a desired, administrative heatup or cooldown rate. Also the Pressurizer administrative heatup rate has been identified as an unnecessary delay while drawing a steam bubble following the vacuum refill process. The time that the Reactor Coolant System is under a vacuum condition will be shortened, resulting in a slight improvement in outage safety. The revision to OP/1(2)/A/6100/01 and associated procedures will establish administrative limits in these procedures as a consistent percentage of the licensing basis limits. Improved Technical Specification (ITS) 3.4.3 limits apply to the Reactor Coolant System and Selected Licensee Commitment (SLC) 16.5-4 limits apply to the Pressurizer. This change is an enhancement to outage safety and scheduling. It may also help the operators remember the limits more easily since they will only need to remember the technical specification limit and apply the percentage.

The following changes will be made:

1. The ITS 3.4.3 Limit for the Reactor Coolant System heatup rate is less than 60 degrees F./hr. per Figure 3.4.3-1. The Administrative Limit will be changed from less than 30 degrees F./hr. to less than 50 degrees F./hr.
2. The ITS 3.4.3 Limit for the Reactor Coolant System cooldown rate is less than 100 degrees F./hr. per Figure 3.4.3-2. The Administrative Limit will be changed from less than 75 degrees F./hr. to less than 80 degrees F./hr.
3. The SLC 16.5-4 Limit for Pressurizer heatup rate is less than 100 degrees F./hr. The Administrative Limit will be changed from less than 50 degrees F./hr. to less than 80 degrees F./hr.
4. The SLC 16.5-4 Limit for Pressurizer cooldown rate is less than 200 degrees F./hr. The Administrative Limit will be changed from less than 100 degrees F./hr. to less than 160 degrees F./hr.

The proposed changes are confined to only the temperature limits shown above. The existing limitations on boltup temperature and minimum temperature for the number of Reactor Coolant Pumps in operation (due to Low Temperature Over Pressure limitations per ITS 3.4.12) remain in effect. Research into the Catawba Improved Technical Specifications, UFSAR, and Safety Evaluation Report, as well as the Design Basis Specification, and supporting calculations has been conducted. Other than the specific mention of the Reactor Coolant System "expected normal rate" of 50 degrees F./hr. in Section 3.9. 1.1 and the "plant operating limits of 75 degrees F./hr. for normal operations" in section 5.3.3.1, all other documents and analyses reference the "heatup and cooldown at 100 degrees F./hr. design case" which is presented in Section 3.9.1.1. Current ITS limits the Reactor Coolant System heatup rate to less than 60 degrees F./hr. per Figure 3.4.3-1 but recent test specimen analyses presented in Westinghouse WCAP-15118 indicate acceptable reactor vessel service life is justifiable at a heatup rate of less than 100 degrees F./hr. (future change to ITS required if this is justified). Chapter 15, Item 3 "Operational Transients" cites the Technical Specification values for all four Reactor Coolant System and pressurizer heatup and cooldown limits. The SER relies solely on the materials data supplied for the Unit 1 and Unit 2 reactor vessels, as well as the most limiting accident transients to evaluate the adequacy of reactor pressure vessel and pressure boundary piping (i.e. there are no normal heatup or cooldown rates discussed or committed in the SER).

Per UFSAR Section 3.9.1.1, 100 degrees F./hr. is stated as the normal cooldown and heatup rate. Supporting calculations show this to be the case. One calculation used a "normal cooldown" rate of 50 degrees F./hr., but further investigation revealed that this was a model of the pressurizer auxiliary spray induced transient following the then current administratively controlled 50 degrees F./hr. cooldown. The 50 degrees F./hr. rate was not significant, rather, the cold spray was the only portion analyzed for stresses. Raising the administrative limit to 80 degrees F./hr. is of no consequence to the stresses imposed by such cold spray (it affects the time prior to the transient of interest). Since the proposed change to 80 degrees F./hr. is less than the 100 degrees F./hr. cooldown transients for which the Reactor Coolant System pressure boundary is qualified, this change is within the plant design basis.

The Westinghouse Outage Optimization Study recommended that the administrative cooldown rate be increased. The 10CFR50.59 Evaluation to justify the subject change to the cooldown rate from 50 degrees F./hr. to 75 degrees F./hr. made reference to the engineering design basis calculations which already assume a plant lifetime of Technical Specification cooldowns and heatups at the full Technical Specification limits. The additional increase from 75 to 80 degrees F./hr. is inconsequential with respect to plant transient analyses and improves human factors in the memorization of administrative limits as approximately 80% of the Catawba ITS and SLC limits. Therefore, there is no reduction in the margin of safety as previously analyzed as long as the Heatup/Cooldown rate is linear and not in step changes. This same logic applies to the other rates as well.

The administrative limit sets a reasonable basis such that if this rate is attempted, it is unlikely that the Technical Specification limit (which is assumed in the plant piping systems lifetime stress analysis) will be exceeded. Operator interpretation can result in unnecessary conservatism. The operator may approach the administrative limit.

However, it is important that the Heatup/Cooldown rate is linear and not in step changes.

Evaluation: This change is not a physical change to the station. It is a change to the normal Unit Startup and Unit Shutdown procedures to allow a higher Reactor Coolant System heatup and cooldown rate which is still within the qualified limits of less than 60 degrees F./hr. (for heatup) and less than 100 degrees F./hr. (for cooldown) imposed by plant Technical Specifications. The normal Unit Startup and Unit Shutdown procedures are also being changed to allow a higher pressurizer heatup and cooldown rate which is still within the qualified limits of less than 100 degrees F./hr. (for heatup) and less than 200 degrees F./hr. (for cooldown) imposed by plant Selected Licensee Commitments. Additionally, it provides justification for the UFSAR change to Sections 3.9.1.1 and 5.3.3.1 in order to reflect the revised "normal" Reactor Coolant System cooldown rate of 80 degrees F./hr. in comparison to the analyzed rate of 100 degrees F./hr.

The accidents described and evaluated in the UFSAR accident analysis are not adversely impacted by increasing the normal Reactor Coolant System heatup rate from 30 to 50 degrees F./hr. and the normal cooldown rate from 75 to 80 degrees F./hr., since the new rate is still bounded by the analyzed heatup and cooldown rate of 100 degrees F./hr. Likewise, the accident analysis is not adversely impacted by increasing the pressurizer heatup rate from 50 to 80 degrees F./hr. and cooldown rate from 100 to 160 degrees F./hr. which is still within the qualified limits of less than 100 degrees F./hr. (for heatup) and less than 200 degrees F./hr. (for cooldown). The primary piping is analyzed for 200 such occurrences, which are tracked and logged as part of the reactor trip assessment process. The performance of components or systems will not be degraded by the change. No equipment used for any phase of either power generation or conversion or transmission, normal shutdown cooling, fuel handling, or radwaste treatment is physically affected. Therefore, the probability of occurrence of an accident previously evaluated in the SAR is not increased.

No system used to mitigate any accident is degraded. The frequency of challenges to equipment provided to mitigate accidents is not increased. The structural qualification of safety related piping has not been degraded. The post fire safe shutdown capability of the plant has not been degraded. Therefore, the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR is not increased.

Neither any fission product barrier nor any source term is adversely affected. The consequences of any accident as described in the SAR would be the same, whether such accidents were to occur at either the old or the new heatup or cooldown rates. Therefore the change will not increase the consequences of an accident previously evaluated in the SAR.

Neither any new failure modes nor any common cause failure modes are created. The proper operation of, as well as the possible failure modes of the Reactor Coolant System and Emergency Core Cooling Systems are not adversely affected by the change in Reactor Coolant System and pressurizer heatup/cooldown limits. Therefore the consequences of a malfunction of equipment important to safety previously evaluated in the SAR are not increased.

The accidents evaluated in the SAR represent a broad spectrum of limiting events. Cooldown rate, if extremely in excess of the analyzed limit of 100 F./hr., may have an adverse effect on return to criticality or pressure boundary integrity. The overcooling events, as well as the Large Break LOCA and Small Break LOCA are already representative of these types of accidents. Therefore the higher allowable administrative cooldown rate does not exceed those already analyzed for these accidents. Per the discussions above, the performance of components or systems will not be degraded by the change.

Neither any new failure modes nor any common cause failure modes are created, and limiting pipe breaks are already analyzed in Large Break LOCA and Small Break LOCA analyses. The possibility of a malfunction of equipment of a different type than any previously evaluated in the SAR is not created.

The margins present in the ITS 3.4.3 Reactor Coolant System Heatup and Cooldown Curves are not affected by the increase in administrative heatup and cooldown rate. These curves (along with LTOP requirements per ITS 3.4.12) impose the necessary conservative limits based on the limiting materials properties of the reactor pressure boundary. While the procedure change from 75 to 80 degrees F./hr. will allow faster Reactor Coolant System cooldown in some portions of the cold shutdown process, and the change from 30 to 50 degrees F./hr. will allow faster heatup when possible, operating procedures ensure that the rate is intended to remain linear as assumed in the analysis, and should not be done in a step change manner.

The margins present in SLC 16.5-7 Pressurizer administrative heatup and cooldown limits are not affected by the increase in administrative heatup and cooldown rate since the pressurizer analysis is done assuming full SLC limits of 100 degrees F./hr. (heatup) and 200 degrees F./hr. (cooldown). The change from an administrative limit of 50 to 80 degrees F./hr. will allow for faster heatup of the pressurizer during the vacuum fill process which has been specifically identified as a barrier to breaking vacuum, thus resulting in a slight improvement in outage safety. Operating procedures ensure that the heatup rate is intended to remain as linear as possible as assumed in the analysis, and not in a step change manner. Pressurizer cooldown has not been specifically identified as a barrier, but changing the administrative limit of 100 to 160 degrees F./hr. will allow faster cooldown in some portions of the cold shutdown when possible, with no effect on accident response or transient stress analysis.

No changes are required to the applicable Technical Specifications. The plant will still be operated in conjunction with of Catawba ITS (Catawba Technical Specification Amendments 173 (Unit 1) and 165 (Unit 2) including the Bases). Therefore, the margin of safety as defined in the bases to any Technical Specification has not been reduced. There are no unreviewed safety questions associated with these changes. UFSAR Section 3.9.1.1 "Design Transients" and 5.3.3.1 "Reactor Vessel Design" will be revised to include the revised administrative limits.

24 Type: Procedure **Unit:** 0

Title: Procedure OP/1(2)/A/6100/02, Controlling Procedure for Unit Shutdown

Description: Procedure OP/1(2)/A/6100/02 "Controlling Procedure for Unit Shutdown", other supporting systems and surveillance procedures, and UFSAR Sections 3.9. 1.1 "Design Transients" and 5.3.3.1, "Reactor Vessel Design" were revised to allow higher administrative limits for Reactor Coolant System and Pressurizer Heatup and Cooldown to be implemented.

This change to the Operating Procedures and corresponding UFSAR sections will implement administrative Heatup and Cooldown limits that are approximately 80% of the licensing basis values.

In the Operating Procedures, administrative heatup and cooldown limits for the Reactor Coolant System and the Pressurizer are generally around 50% of the Technical Specification Limits. Recently, changes for outage optimization were made to one of these limits, (Reactor Coolant System Cooldown) to move the Administrative limit up to 75 degrees F/hour cooldown, or 75% of the Technical Specification Limits. A review of how these "limits" are understood found inconsistencies. Some operators attempt to operate near the limit, while others will operate at about half of these values. This caused difficulty in planning an outage. Through the use of newer technology, the operators are now able to more precisely follow a desired, administrative heatup or cooldown rate. Also the Pressurizer administrative heatup rate has been identified as an unnecessary delay while drawing a steam bubble following the vacuum refill process. The time that the Reactor Coolant System is under a vacuum condition will be shortened, resulting in a slight improvement in outage safety. The revision to OP/1(2)/A/6100/02 and associated procedures will establish administrative limits in these procedures as a consistent percentage of the licensing basis limits. Improved Technical Specification (ITS) 3.4.3 limits apply to the Reactor Coolant System and Selected Licensee Commitment (SLC) 16.5-4 limits apply to the Pressurizer. This change is an enhancement to outage safety and scheduling. It may also help the operators remember the limits for knowledge purposes as they only need to remember the technical specification limit and apply the percentage.

The following changes will be made:

1. The ITS 3.4.3 Limit for the Reactor Coolant System heatup rate is less than 60 degrees F./hr. per Figure 3.4.3-1. The Administrative Limit will be changed from less than 30 degrees F./hr. to less than 50 degrees F/hr.
2. The ITS 3.4.3 Limit for the Reactor Coolant System cooldown rate is less than 100 degrees F./hr. per Figure 3.4.3-2. The Administrative Limit will be changed from less than 75 degrees F./hr. to less than 80 degrees F./hr.
3. The SLC 16.5-4 Limit for Pressurizer heatup rate is less than 100 degrees F./hr. The Administrative Limit will be changed from less than 50 degrees F./hr. to less than 80 degrees F./hr.
4. The SLC 16.5-4 Limit for Pressurizer Cooldown rate is less than 200 degrees F./hr. The Administrative Limit will be changed from less than 100 degrees F./hr. to less than 160 degrees F./hr.

The proposed changes are confined to only the temperature limits shown above. The existing limitations on boltup temperature and minimum temperature for the number of Reactor Coolant Pumps in operation (due to Low Temperature Over Pressure limitations per ITS 3.4.12) remain in effect. Research into the Catawba Improved Technical Specifications, UFSAR, and Safety Evaluation Report, as well as the Design Basis Specification, and supporting calculations has been conducted. Other than the specific mention of the Reactor Coolant System "expected normal rate" of 50 degrees F./hr. in Section 3.9. 1.1 and the "plant operating limits of 75 degrees F./hr. for normal operations" in section 5.3.3. 1, all other documents and analyses reference the "heatup and cooldown at 100 degrees F./hr. design case" which is presented in Section 3.9.1.1. Current ITS limits the Reactor Coolant System heatup rate to less than 60 degrees F./hr. per Figure 3.4.3-1 but recent test specimen analyses presented in Westinghouse WCAP-15118 indicate acceptable reactor vessel service life is justifiable at a heatup rate of less than 100 degrees F./hr. (future change to ITS required if this is justified). Chapter 15, Item 3 "Operational Transients" cites the Technical Specification values for all four Reactor Coolant System and pressurizer heatup and cooldown limits. The SER relies solely on the materials data supplied for the Unit 1 and Unit 2 reactor vessels, as well as the most limiting accident transients to evaluate the adequacy of reactor pressure vessel and pressure boundary piping (i.e. there are no normal heatup or cooldown rates discussed or committed in the SER).

Per UFSAR Section 3.9.1.1, 100 degrees F./hr. is stated as the normal cooldown and heatup rate. Supporting calculations show this to be the case. One calculation used a "normal cooldown" rate of 50 degrees F./hr., but further investigation revealed that this was a model of the pressurizer auxiliary spray induced transient following the then current administratively controlled 50 degrees F./hr. cooldown. The 50 degrees F./hr. rate was not significant, rather, the cold spray was the only portion analyzed for stresses. Raising the administrative limit to 80 degrees F./hr. is of no consequence to the stresses imposed by such cold spray (it affects the time prior to the transient of interest). Since the proposed change to 80 degrees F./hr. is less than the 100 degrees F./hr. cooldown transients for which the Reactor Coolant System pressure boundary is qualified, this change is within the plant design basis.

The Westinghouse Outage Optimization Study recommended that the administrative cooldown rate be increased. The 10CFR50.59 Evaluation to justify the subject change to the cooldown rate from 50 degrees F./hr. to 75 degrees F./hr. made reference to the engineering design basis calculations which already assume a plant lifetime of Tech Spec cooldowns and heatups at the full Tech Spec limits. The additional increase from 75 to 80 degrees F./hr. is inconsequential with respect to plant transient analyses and improves human factors in the memorization of Administrative limits as approximately 80% of the Catawba ITS and SLC limits. Therefore, there is no reduction in the margin of safety as previously analyzed as long as the Heatup/Cooldown rate is linear and not in step changes. This same logic applies to the other rates as well.

The Administrative limit sets a reasonable basis such that if this rate is attempted, it is unlikely that the Technical Specification limit (which is assumed in the plant piping systems lifetime stress analysis) will be exceeded. Operator interpretation can result in unnecessary conservatism. The operator may approach the Administrative limit. However, it is important that the Heatup/Cooldown rate is linear and not in step changes.

Evaluation: This change is not a physical change to the station. It is a change to the normal Unit Startup and Unit Shutdown procedures to allow a higher Reactor Coolant System heatup and cooldown rate which is still within the qualified limits of less than 60 degrees F./hr. (for heatup) and less than 100 degrees F./hr. (for cooldown) imposed by plant Technical Specifications. Likewise, the normal Unit Startup and Unit Shutdown procedures are also being changed to allow a higher pressurizer heatup and cooldown rate which is still within the qualified limits of less than 100 degrees F./hr. (for heatup) and less than 200 degrees F./hr. (for cooldown) imposed by plant Selected Licensee Commitments. Additionally, it provides justification for the UFSAR change to Sections 3.9.1.1 and 5.3.3.1 in order to reflect the revised "normal" Reactor Coolant System cooldown rate of 80 degrees F./hr. in comparison to the analyzed rate of 100 degrees F./hr.

The accidents described and evaluated in the UFSAR accident analysis are not adversely impacted by increasing the normal NC System heatup rate from 30 to 50 degrees F./hr. and the normal cooldown rate from 75 to 80 degrees F./hr., since the new rate is still bounded by the analyzed heatup and cooldown rate of 100 degrees F./hr. Likewise, the accident analysis is not adversely impacted by increasing the pressurizer heatup rate from 50 to 80 degrees F./hr. and cooldown rate from 100 to 160 degrees F./hr. which is still within the qualified limits of less than 100 degrees F./hr. (for heatup) and less than 200 degrees F./hr. (for cooldown). The primary piping is analyzed for 200 such occurrences, which are tracked and logged as part of the reactor trip assessment process. The performance of components or systems will not be degraded by the change. No equipment used for any phase of either power generation or conversion or transmission, normal shutdown cooling, fuel handling, or radwaste treatment is physically affected. Therefore, the probability of occurrence of an accident previously evaluated in the SAR is not increased.

No system used to mitigate any accident is degraded. The frequency of challenges to equipment provided to mitigate accidents is not increased. The structural qualification of safety related piping has not been degraded. The post fire safe shutdown capability of the plant has not been degraded. Therefore, the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR is not increased.

Neither any fission product barrier nor any source term is adversely affected. The consequences of any accident as described in the SAR would be the same, whether such accidents were to occur at either the old or the new heatup or cooldown rates. Therefore the change will not increase the consequences of an accident previously evaluated in the SAR.

Neither any new failure modes nor any common cause failure modes are created. The proper operation of, as well as the possible failure modes of the Reactor Coolant System and Emergency Core Cooling Systems are not adversely affected by the change in Reactor Coolant System and pressurizer heatup/cooldown limits. Therefore the consequences of a malfunction of equipment important to safety previously evaluated in the SAR are not increased.

The accidents evaluated in the SAR represent a broad spectrum of limiting events.

Cooldown rate, if extremely in excess of the analyzed limit of 100 F./hr., may have an adverse effect on return to criticality or pressure boundary integrity. The overcooling events, as well as the Large Break LOCA and Small Break LOCA are already representative of these types of accidents. Therefore the higher allowable administrative cooldown rate does not exceed those already analyzed for these accidents. Per the discussions above, the performance of components or systems will not be degraded by the change.

Neither any new failure modes nor any common cause failure modes are created, and limiting pipe breaks are already analyzed in Large Break LOCA and Small Break LOCA analyses. The possibility of a malfunction of equipment of a different type than any previously evaluated in the SAR is not created.

The margins present in the ITS 3.4.3 Reactor Coolant System Heatup and Cooldown Curves are not affected by the increase in administrative heatup and cooldown rate. These curves (along with LTOP requirements per ITS 3.4.12) impose the necessary conservative limits based on the limiting materials properties of the reactor pressure boundary. While the procedure change from 75 to 80 degrees F./hr. will allow faster Reactor Coolant System cooldown in some portions of the cold shutdown process, and the change from 30 to 50 degrees F./hr. will allow faster heatup when possible, operating procedures ensure that the rate is intended to remain linear as assumed in the analysis, and should not be done in a step change manner.

The margins present in SLC 16.5-7 Pressurizer administrative heatup and cooldown limits are not affected by the increase in administrative heatup and cooldown rate since the pressurizer analysis is done assuming full SLC limits of 100 degrees F./hr. (heatup) and 200 degrees F./hr. (cooldown). The change from an administrative limit of 50 to 80 degrees F./hr. will allow for faster heatup of the pressurizer during the vacuum fill process which has been specifically identified as a barrier to breaking vacuum, thus resulting in a slight improvement in outage safety. Operating procedures ensure that the heatup rate is intended to remain as linear as possible as assumed in the analysis, and not in a step change manner. Pressurizer cooldown has not been specifically identified as a barrier, but changing the administrative limit of 100 to 160 degrees F./hr. will allow faster cooldown in some portions of the cold shutdown when possible, with no effect on accident response or transient stress analysis.

No changes are required to the applicable Technical Specifications. The plant will still be operated in conjunction with of Catawba ITS (Catawba Technical Specification Amendments 173 (Unit 1) and 165 (Unit 2) including the Bases). Therefore, the margin of safety as defined in the bases to any Technical Specification has not been reduced. There are no unreviewed safety questions associated with these changes. UFSAR Section 3.9.1.1 "Design Transients" and 5.3.3.1 "Reactor Vessel Design" will be revised to include the revised administrative limits.

21 Type: Procedure **Unit:** 1

Title: Procedure OP/1/A/6200/004, Enclosure 4.14, Venting Residual Heat Removal System Discharge Headers

Description: Steps 2.3 and 2.4 of Enclosure 4.14 of procedure OP/l/A/6200/004 were deleted. Steps 2.9 and 2.10 were deleted as well. These steps addressed the removal/installation of the pipe cap/tubing from valve 1ND30. The pipe cap is now permanently removed and the tubing permanently installed so these steps are not needed. A new procedure step 2.4 was added to ensure the venting connections are in place.

The removal of the pipe cap and flexible tubing installation was performed per a work order and was exempted from the modification process in accordance with CNS Site Directive 4.4.5, Section 5.2.4. The procedure change was necessary to delete steps specifically controlling the piping configuration during the performance of the procedure. The permanent changes provided by the work order eliminate the need for these steps in the procedure.

Evaluation: The only affected system, structure or component is the pipe cap associated with the downstream non-nuclear safety related piping of vent valve 1ND30. The only pertinent, related design event associated with this system, structure or component is a seismic event which may potentially affect the upstream piping which is nuclear safety related. No new or different failure modes are created for this component resulting from the removal of this pipe cap. No reanalysis of an accident or design basis is necessary resulting from this procedure change.
The changes made to Enclosure 4.14 of OP/1/A/6200/004 reflect the permanent removal of the pipe cap associated with work order task 98125136 01. The removal of these steps within the procedure relate the procedure to an approved design configuration of non-nuclear safety related piping downstream of valve 1ND30. The probability of occurrence of an accident is not increased based on this change since the removed pipe cap is not nuclear safety related and does not perform a safety function. This procedure change will not prevent any system, structure or component from performing its design function. There are no unreviewed safety questions associated with this procedure change. No Technical Specification changes are required. No UFSAR changes are required.

232 **Type:** Procedure **Unit:** 0

Title: Procedure OP/1/A/6200/009 Revision 64B and OP/2/A/6200/009 Revision 54B

Description: Procedure OP/1/A/6200/009 Revision 64B and OP/2/A/6200/009 Revision 54B, "Cold Leg Accumulator Operation", Enclosure 4.5 , "Decreasing Cold Leg Accumulator Level", is being revised to allow an alternate method of decreasing Cold Leg Accumulator level by establishing a flow path through the sample lines. This will be the preferred method of decreasing the Cold Leg Accumulator level. This method is preferred because it decreases the potential for unseating the Cold Leg Accumulator fill and Cold Leg Accumulator check valve test isolation valves that can result in increased in-leakage to the Cold Leg Accumulator. Previously Cold Leg Accumulator level was decreased through the Safety Injection System test header off the Cold Leg Accumulator discharge line.

Evaluation: There are no unreviewed safety questions associated with this alternate drain configuration. Methods for determining boron concentration are adequate to ensure the boron concentration is maintained within operability limits whether the Cold Leg Accumulator level decrease is established through sample lines or through the Safety Injection System test header. No Technical Specification changes are required. No UFSAR changes are required. The Cold Leg Accumulator functions along with other Emergency Core Cooling System flows , to ensure that maximum fuel cladding temperature, cladding oxidation and hydrogen generation limits are not exceeded and the reactor core is maintained in a coolable geometry following a LOCA. This change does not affect the boron concentration, volume, or pressure in the accumulators. Therefore, the assumptions of the LOCA analysis are not affected by this change. This change does not decrease the margin of safety for the Cold Leg Accumulators and no change to the fission product barriers from the assumptions of the UFSAR will result from this change.

165 **Type:** Procedure **Unit:** 0

Title: Procedure OP/1/A/6200/05 Revision 66 and OP/2/A/6200/05 Revision 51 (Unit 1, 2 Spent Fuel Cooling System)

Description: Procedure changes OP/1/A/6200/05 Revision 66 and OP/2/A/6200/05 Revision 51 (Unit 1, 2 Spent Fuel Cooling System) will add an enclosure to allow the use of one Spent Fuel Cooling System Pump flowing through both Heat Exchnagers. This alignment will maximize the heat transfer ability of the Spent Fuel Cooling System when only one pump is available to run. These procedure changes will also modify the enclosures for placing the purification loop in service and removing the purification loop from service to properly interface with the new dual heat exchanger enclosure being added.

Evaluation: The Spent Fuel Cooling System is normally in service at all times. One Spent Fuel Pump and one Spent Fuel Heat Exchanger are the normal components in service at one time. The system is shutdown during a safety injection. Decay heat in the Spent Fuel Pool is removed by boiling with assured makeup water from the Nuclear Service Water System. The Spent Fuel Cooling System does not serve any safety related function during a design basis accident. The pumps are supplied with Class 1E power from the diesel generators which can be used during a blackout. The pumps must be manually started during a blackout. There are infrequent times in which there is only one pump available but two heat exchangers are available. In order to maximize the heat removal capability of the Spent Fuel Cooling Stystem during these times, it was desired to be able to place flow from a single pump through both heat exchangers. A new procedure enclosure was written for each unit that aligns both Spent Fuel Pool Heat Exchangers in parallel with the pump. The enclosure was written as guidance to ensure that the valve lineup is performed in the correct sequence, pump parameters are kept in the nominal range, and the purification loop is operated correctly in the new alignment. The new enclosure will also provide directions on how to realign back to a normal two train alignment. There are no unreviewed safety questions associated with this procedure change . The Spent Fuel cooling system is not an accident initiator and plays no role in any of the accidents analyzed in the UFSAR, 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.

84 Type: Procedure **Unit:** 1

Title: Procedure OP/1/A/6200/11, Revision 36, Operating Procedure for the Primary Sample System

Description: Enclosure 4.35 is being added to Procedure OP/1/A/6200/11 to allow degassing the pressurizer steam space to the Nuclear Sampling System sample hood during plant shutdown.

During the reducing phase of crudburst at plant shutdown, it is important to manage dissolved hydrogen concentration. Sufficient hydrogen maintains a reducing condition to optimally decompose nickel ferrite for subsequent solublization and cleanup during the oxidative phase of crudburst. Maintaining a high hydrogen residual is preferred but challenges the ability to degas in reasonable timeframe for oxidative crudburst. Oxidative crudburst requires < 5 cc/kg dissolved hydrogen. Degassing hydrogen from the pressurizer steam space to the sample hood ensures good control over hydrogen concentration in tandem with controlled use of standard Waste Gas System degassing of the Volume Control Tank (VCT). This method will also be used to vent the pressurizer during extended mid-cycle outages in Modes 3-5, to prevent high hydrogen concentration in the reactor coolant. Venting the pressurizer steam space through the Nuclear Sampling System to the VCT gas space has historically yielded poor results.

This activity is being performed to vent hydrogen from the pressurizer steam space through the Nuclear Sampling System sample line. Since hydrogen will be purged, flammability concerns have been addressed in a Memorandum to File, dated Oct. 6, 1997. The applicable sections are for normal operating conditions and worst case of venting 100% hydrogen. As shown, dilution from sample hood vent flow brings the concentration far below the 4% flammability limit at the sample hood discharge.

The actual activity of purging or venting the pressurizer steam space is equivalent to an extended sample purge. It would be the same as adding an additional 10-15 hours of sampling to the Nuclear Sampling System on an annual basis. The activity is identical to sampling, with the exception of actually obtaining a sample, and requires no modifications to the Nuclear Sampling System. Sampling in itself is a process, which vents or drains the system from which sampling is being performed. Supplemental use of the Nuclear Sampling System for venting or draining does not challenge the design basis of the Nuclear Sampling System. Its use for this purpose is stated in EPRI-TR-1 05714 p. B-1 1 and has been standard industry practice. A corollary would be the industry practice of using vent and drain lines as alternative sample points. Obtaining a sample during a vent or drain activity would also not challenge any design basis.

Reactor Coolant System, pressurizer, pressurizer steam, accumulators, etc (sample lines with origins inside containment) have containment isolation valves which fail in the closed position upon receiving a Phase A Containment Isolation Signal. Lines outside of containment such as those of the Residual Heat Removal System would be controlled procedurally or through the use of compensatory actions. These measures would control release of radioactive materials to the environment.

As with any sampling of a primary system in the Nuclear Sampling System sample hood, activity is released to the auxiliary building filtered ventilation system. Both this system and the plant vent stack, is monitored by radiation monitors. The activity of venting the pressurizer steam space will be assessed procedurally by calculating the curies released from the sample hood during the activity. Radiation Protection will then use this data in the Auxiliary Building monthly accountability package. As an initial condition, reactor coolant activity will be evaluated by Chemistry and Radiation Protection prior to the outage, to assess the pressurizer venting. Radiation Protection will also be contacted prior to the purge to alert them to monitor for any unusual activity at the Auxiliary Building monitor or plant vent stack. Chemistry will terminate the purge at any time as Radiation Protection dictates.

Reactivity concerns are not an issue during this activity as removing any steam from the pressurizer would cause a small increase in boron in the pressurizer. Since the boron in the pressurizer is lagging behind the reactor coolant during this phase of shutdown, this actually aids the boration and also aids in the prevention of the pressurizer being a dilution source for reactor coolant.

The performance of this enclosure will be restricted to Modes 3-6 during plant shutdown. During normal operation, Mode 2, or plant startup the activity in the pressurizer steam space is marginal since the volume is essentially all steam with marginal gas.

The title of Enclosure 4.34, Pressurizer Steam Space Alignment To Sample Hood will be changed to "Pressurizer Steam Space Alignment To Sample Hood During Plant Startup" to be more specific in differentiating it from this new enclosure.

Evaluation: There are no unreviewed safety questions associated with this procedure revision. This procedure describes the use of the Nuclear Sampling System to remove hydrogen from the Pressurizer during plant shutdown. The activity during the performance of this procedure is the same as a sample purge prior to obtaining a sample. The routing of high pressure and temperature sample lines outside containment is not considered hazardous because of the limited flow capacity.

This activity utilizes the as built Nuclear Sampling System. The use of this system would not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR since there is no direct link to safety related equipment.

The Nuclear Sampling System being utilized by this activity will automatically isolate on a Phase A Containment Isolation Signal, thus terminating any venting in progress.

The Chemical and Volume Control System is isolated from equipment important to safety previously evaluated in the SAR. The operation of the Nuclear Sampling System would not then increase the consequences of a malfunction of safety related equipment.

Since the activity described in this procedure is identical to the process for purging a sample, any accident scenario would also encompass normal sampling. The routing of high pressure and temperature sample lines outside containment is not considered hazardous because of the limited flow capacity.

The Nuclear Sampling System is isolated from equipment important to safety previously evaluated in the SAR. The operation of the System would not create the possibility of a malfunction of equipment important to safety other than any evaluated previously in the SAR.

The Nuclear Sampling System being utilized by this activity will automatically isolate on a Phase A Containment Isolation Signal, thus terminating any venting in progress. Removal of steam from the pressurizer gas space will actually aid in discounting the pressurizer as a dilution source when achieving Mode 6 concentration for boron.

This procedure allows for proper control of plant chemistry during shutdown. Chemistry actions for venting the pressurizer steam space are described. The changes are in full compliance with Technical Specifications, Selected Licensee Commitments, Core Operating Limits Report, and the SAR. No Technical Specification changes are required. No UFSAR changes are required. Since the process of obtaining a sample is a venting or draining activity, the supplemental use of the to Nuclear Sampling System to support venting or draining does not challenge the design basis of the system. Any activity released from the sample hood during this activity will be accounted for and controlled procedurally.

48 **Type:** Procedure **Unit:** 1

Title: Procedure OP/1/A/6200/28, "Operating Procedure for the Addition of Chemicals to the Reactor Coolant System".

Description: Procedure OP/1/A/6200/28, "Operating Procedure for the Addition of Chemicals to the Reactor Coolant System", was revised to add an enclosure which addresses injection of hydrazine into the Residual Heat Removal System. Hydrazine will be injected into the Residual Heat Removal System piping in sufficient quantity to scavenge the residual oxygen and maintain a slight reducing environment. The hydrazine should be injected prior to shutdown in a time frame to allow for the hydrazine to react with the dissolved oxygen at the low Residual Heat Removal System temperature. The hydrazine will be injected into the suction side of the Residual Heat Removal pump via a drain valve. An approximately equivalent amount of water will be let down from the sample line (or equivalent valve) to prevent Residual Heat Removal System pressurization. Operations will then perform their standard Operating Procedure to re-circulate the Residual Heat Removal System and provide mixing of the hydrazine. Hydrazine will be added as 32% hydrazine, which has no flash point.

Evaluation: Excessive flow margin of 200 gallons per minute in the Residual Heat Removal System flow accident analysis provides assurance that injection requirements are met until Operations and Chemistry can remotely close the Residual Heat Removal System pump drain valve and sample line. Closing the drain and sample lines prior to manual realignment to long term sump recirculation ensures that control room and offsite dose is not increased above previous values. The fission product barriers of the fuel pellet, clad, reactor coolant system primary pressure boundary and containment are not adversely affected as a result of this procedure. No Technical Specification changes are required. There are no Unreviewed Safety Questions associated with this procedure revision. No UFSAR changes are required.

243 Type: Procedure **Unit:** 1

Title: Procedure OP/1/A/6250/002 Revision 106B, Auxiliary Feedwater System

Description: Procedure OP/1/A/6250/002, Auxiliary Feedwater System, is being revised to add Enclosure 4.19, "Operability Concerns with Nuclear Service Water System to Auxiliary Feedwater System Assured Makeup Line Degraded." This enclosure provides guidance to Operations should Engineering determine that the line is degraded per the acceptance criteria in PT/1/A/4400/014 "Nuclear Service Water System to Auxiliary Feedwater System Suction Piping Flow Measurement Test." Procedure PT/1/A/4400/014 has curves for Nuclear Service Water System flow versus differential pressure for one Nuclear Service Water Pump and for two Nuclear Service Water Pumps. If the test results fall below the "one pump" curve only one Nuclear Service Water Pump is required to ensure the Auxiliary Feedwater System pump pressure remains positive and no action is required by Engineering or Operations. If the test results fall between the "one pump" and "two pump" curves, two Nuclear Service Water Pumps are required to ensure the Auxiliary Feedwater Pump suction pressure remains positive. In this case, Engineering notifies Operations of the test results. The enclosure then directs Operations to determine if two Nuclear Service Water Pumps are operable on the affected train. If so, the Train related Auxiliary Feedwater Pump is operable. If only one Nuclear Service Water Pump is operable (on the same train as the Auxiliary Feedwater Train that was tested), Operations declares the associated Motor Driven Auxiliary Feedwater Pump inoperable. Also, if the test results fall above the two pump curve, the Enclosure provides guidance to declare the train related Auxiliary Feedwater Pump inoperable. This would be due to the inability of the associated Nuclear Service Water train to supply adequate pressure to ensure that the Auxiliary Feedwater Pump suction pressure remains positive. The Auxiliary Feedwater Turbine Driven Pump remains operable since the opposite train, non-degraded Nuclear Service Water train can supply 100% of the Auxiliary Feedwater Turbine Driven Pump demand.

Evaluation: There are no unreviewed safety questions associated with this procedure revision. Neither the Nuclear Service Water System nor the Auxiliary Feedwater System are initiators of accidents analyzed in the UFSAR. The procedure change adds guidance on Auxiliary Feedwater System operability based on the condition of the Nuclear Service Water System to Auxiliary Feedwater System assured makeup piping. No Technical Specification changes are required. No UFSAR revisions are required.

231 **Type:** Procedure **Unit:** 2

Title: Procedure OP/2/A/6100/001, Revision 120 "Controlling Procedure for Unit Startup", OP/2/A/6100/002, Rev 120, "Controlling Procedure for Unit Shutdown, and OP/2/A/6100/005, Rev 57, "Unit Fast Recovery".

Description: Valve 2CA-223, the manual isolation valve for individual tempering flow to Steam Generator 2A cannot be opened, leaving Steam Generator 2A without tempering flow. An evaluation has concluded that with no tempering flow to Steam Generator 2A, the cooldown transient caused by the swapover of flow from the Main Feedwater Nozzle to the Auxiliary Feedwater Nozzle at approximately 15% flow should be counted as equivalent to the transient "Auxiliary Feedwater Actuation without Tempering Flow:" Plant downpower and recovery evolutions will now be captured as applicable transients when they result in loss of Main Feedwater flow to the Steam Generator 2A Auxiliary Feedwater nozzle without tempering flow in service.

Procedures OP/2/A/6100/001, "Controlling Procedure for Unit Startup", OP/2/A/6100/002, "Controlling Procedure for Unit Shutdown, and OP/2/A/6100/005, "Unit Fast Recovery", currently close valve 2CA223 to align the Main Feedwater and Auxiliary Feedwater Systems for reverse purge and subsequently reopen the valve to reestablish tempering flow prior to transferring Main Feedwater flow to the Main Feedwater nozzle. The procedures will be revised to require the operator to verify that valve 2CA223 is closed rather than to close the valve and inform the control room SRO that valve 2CA223 cannot be opened where the procedures previously required the valve to be open.

Evaluation: The revisions to these procedures do not introduce an unreviewed safety question. The procedure revisions provide instructions to plant operators on how to address the present failed closed position of valve 2CA223. These instructions ensure that plant operation with this valve out of service will not adversely affect the fatigue life of the Steam Generator 2A auxiliary feedwater nozzle, the failure of which would result in this Steam Generator being in a faulted condition. These procedure changes ensure that this failure will not occur as a result of the condition of valve 2CA223. No Technical Specification changes are required. No UFSAR changes are required.

244 **Type:** Procedure **Unit:** 2

Title: Procedure OP/2/A/6250/002 Revision 96B, Auxiliary Feedwater System

Description: Procedure OP/2/A/6250/002, Auxiliary Feedwater System, is being revised to add Enclosure 4.19, "Operability Concerns with Nuclear Service Water System to Auxiliary Feedwater System Assured Makeup Line Degraded." This enclosure provides guidance to Operations should Engineering determine that the line is degraded per the acceptance criteria in PT/2/A/4400/014 "Nuclear Service Water System to Auxiliary Feedwater System Suction Piping Flow Measurement Test." Procedure PT/2/A/4400/014 has curves for Nuclear Service Water System flow versus differential pressure for one Nuclear Service Water Pump and for two Nuclear Service Water Pumps. If the test results fall below the "one pump" curve only one Nuclear Service Water Pump is required to ensure the Auxiliary Feedwater System Pump pressure remains positive and no action is required by Engineering or Operations. If the test results fall between the "one pump" and "two pump" curves, two Nuclear Service Water pumps are required to ensure the Auxiliary Feedwater Pump suction pressure remains positive. In this case, Engineering notifies Operations of the test results. The enclosure then directs Operations to determine if two Nuclear Service Water Pumps are operable on the affected train. If so, the Train related Auxiliary Feedwater Pump is operable. If only one Nuclear Service Water Pump is operable (on the same train as the Auxiliary Feedwater Train that was tested), Operations declares the associated Motor Driven Auxiliary Feedwater Pump inoperable. Also, if the test results fall above the two pump curve, the Enclosure provides guidance to declare the train related Auxiliary Feedwater Pump inoperable. This would be due to the inability of the associated Nuclear Service Water train to supply adequate pressure to ensure that the Auxiliary Feedwater Pump suction pressure remains positive. The Auxiliary Feedwater Turbine Driven Pump remains operable since the opposite train, non-degraded Nuclear Service Water train can supply 100% of the Auxiliary Feedwater Turbine Driven Pump demand.

Evaluation: There are no unreviewed safety questions associated with this procedure revision. Neither the Nuclear Service Water System nor the Auxiliary Feedwater System are initiators of accidents analyzed in the UFSAR. The procedure change adds guidance on Auxiliary Feedwater System operability based on the condition of the Nuclear Service Water System to Auxiliary Feedwater System assured makeup piping. No Technical Specification changes are required. No UFSAR revisions are required.

126 **Type:** Procedure **Unit:** 0

Title: Procedure PT/0/A/4400/08A, Revison 34 Change A, "Nuclear Service Water System Flow Balance Train A"

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 Containment Spray Heat Exchanger and Control Room Chiller Condenser. Nuclear Service Water 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 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. As part of the flow balance, Nuclear Service Water System header pressure is verified to be adequate to provide makeup to the Containment Seal Water Injection and Auxiliary Feedwater Systems. This restricted change to the test procedure does not alter the purpose of the test or the method of performing the test.

The restricted change will record the Nuclear Service Water System essential header pressure while simulating certain Nuclear Service Water System to Auxiliary Feedwater makeup flow rates, and with one or two Nuclear Service Water System pumps running on the train. The information that will be recorded is for information only and is not part of the flow balance test acceptance criteria. This change does not alter the current flow balance throttle valve positions or affect the test acceptance criteria. The change does not place the Nuclear Service Water System in any abnormal alignments since the system design is to operate with one or two pumps in service on a train.

Evaluation: All of the alignments specified in the change are allowed by Nuclear Service Water System operating procedures. Nuclear Service Water System design allows operation with one or two pumps in service on a train and various flowrates through the Containment Spray System Heat Exchanger. This change does not alter the current flow balance throttle valve positions or affect the test acceptance criteria, therefore the change will not increase the probability of any accident. There are no unreviewed safety questions associated with this procedure revision. No Technical Specification changes are required. No UFSAR changes are required.

127 **Type:** Procedure **Unit:** 0

Title: Procedure PT/0/A/4400/08B, Revison 30 Change A, "Nuclear Service Water System Flow Balance Train 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 Containment Spray Heat Exchanger and Control Room Chiller Condenser. Nuclear Service Water 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 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. As part of the flow balance, Nuclear Service Water System header pressure is verified to be adequate to provide makeup to the Containment Seal Water Injection and Auxiliary Feedwater Systems. This restricted change to the test procedure does not alter the purpose of the test or the method of performing the test.

The restricted change will record the Nuclear Service Water System essential header pressure while simulating certain Nuclear Service Water System to Auxiliary Feedwater makeup flow rates, and with one or two Nuclear Service Water System pumps running on the train. The information that will be recorded is for information only and is not part of the flow balance test acceptance criteria. This change does not alter the current flow balance throttle valve positions or affect the test acceptance criteria. The change does not place the Nuclear Service Water System in any abnormal alignments since the system design is to operate with one or two pumps in service on a train.

Evaluation: All of the alignments specified in the change are allowed by Nuclear Service Water System operating procedures. Nuclear Service Water System design allows operation with one or two pumps in service on a train and various flowrates through the Containment Spray System Heat Exchanger. This change does not alter the current flow balance throttle valve positions or affect the test acceptance criteria, therefore the change will not increase the probability of any accident. There are no unreviewed safety questions associated with this procedure revision. No Technical Specification changes are required. No UFSAR changes are required.

105 Type: Procedure **Unit:** 0

Title: Procedure PT/0/A/4450/020, Revision 1, Ventilation Filter Testing Program

Description: Procedure PT/0/A/4450/020 controls the Ventilation Filter Testing Program. This program was created to satisfy a requirement of the Improved Technical Specifications (ITS) to establish a ventilation filter testing program. While such a program already existed, the new ITS requirement provided an opportunity to clarify several filter testing issues that were subject to interpretation. Therefore a new document, the Ventilation Filter Testing Program, was created to provide guidelines for performing inspections and surveillance testing of the Engineered Safety Features ventilation filter units. Revision 1 of this document made several changes and clarifications to the original procedure. These changes are considered editorial.

Evaluation: There are no unreviewed safety questions associated with this procedure revision. The changes to the procedure consisted of the addition and revision of notes and references. Clarification of two examples were made. These examples were associated with "maximum paintable surfaces" and "smoke volume" and how these parameters affect the Auxiliary Building Ventilation System. None of the changes have any effect on the probability or consequences of accidents evaluated in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required.

253 Type: Procedure **Unit:** 0

Title: Procedure PT/1(2)/A/4200/009 Revision 140T/166L, ESF Actuation Periodic Test Procedure

Description: Procedure PT/1(2)/A/4200/009 Revision 140T (Unit 1) /166L (Unit 2) are being revised to make the following changes. Stroke times for valves 1SV001, 1SV007, 1SV013, 1SV019, 2SV001, 2SV007, 2SV013, 2SV019 are being changed to 8 seconds (9.2 seconds ESF). These valves are the Steam Generator Power Operated Relief Valves. Several different sources list stroke times for these valves with the most conservative value being 8 seconds.

Evaluation: There are no unreviewed safety questions associated with this procedure revision. These changes to the ESF Actuation Periodic Test Procedure do not create any new failure modes or operating characteristics. All testing is performed by approved procedures. The new stroke times are conservative compared to licensing documents. This procedure revision 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.

151 **Type:** Procedure **Unit:** 0

Title: Procedure PT/1(2)/A/4200/09 Revision 166D/140R

Description: A concern was identified with the Containment Air Release and Addition System containment isolation valve ESF Response Times. The ESF Response Times in PT/1(2)/A/4200/09 for valves 1(2)VQ2A, 16A, 3B, and 15B were greater than the six second Purge and Exhaust Isolation Maximum Response Time in UFSAR Table 7-15.

PT/1(2)/A/4200/09, "ESFAS Test", Revisions 165K and 140K, changed the Containment Air Release and Addition System Containment Isolation Valve ESF Response Times from 16.8 (LOCA) and 26.8 (LOCA/Blackout) to 4.8 seconds. These changes were incorrectly evaluated. The valve operators were incorrectly assumed to be air operated. Purge and Exhaust Isolation in UFSAR Table 7-15 (six seconds maximum) was considered to include the Containment Air Release and Addition System containment isolation valves (CIVs), and past test results were not thoroughly reviewed.

The Containment Air Release and Addition System CIV actuators are motor operated. These valves were designed to close on a containment ventilation isolation signal, which includes Safety Injection Phase "A" Isolation, with a maximum isolation (stroke) time of five seconds. Therefore, during a LOCA/Blackout, these valves cannot possibly close within six seconds. ESF LOCA/Blackout Response Times for power operated valves should include maximum delay times of two seconds for SSPS instrumentation, eleven seconds for diesel generator startup and load sequencing, and the design valve isolation or stroke time.

PT/1/A/4200/09 Revision 166D and PT/2/A/4200/09 Revision 140R changed the Containment Air Release and Addition System CIV ESF Response Times back to 16.8 seconds (LOCA) and 26.8 seconds (LOCA/Blackout) .

Evaluation: There are no unreviewed safety questions associated with these procedure changes. No Technical Specification changes are required. UFSAR Table 7-15 was revised to clarify ESF Response Times for the Containment Air Release and Addition System.

166 **Type:** Procedure **Unit:** 1

Title: Procedure PT/1/A/4150/001I, Manual Reactor Coolant System Leakage calculation, revision 1B

Description: Procedure PT/1/A/4150/001I, Manual Reactor Coolant System Leakage Calculation, revision 1B, was prepared July 16, 1999. This procedure is used to perform a manual calculation of Reactor Coolant System leakage in the event that the Operator Aid Computer (OAC) is not available. The procedure substitutes for the function of PT/1/A/4150/001D, Reactor Coolant System Leakage Calculation.

This procedures verifies that identified and unidentified Reactor Coolant System leakage is within the limits specified in ITS 3.4.13b and 3.4.13c. The Reactor Coolant System leakage limits are less than 1 gallon per minute (gpm) unidentified and less than 10 gpm identified leakage.

Also, this procedure verifies that total accumulative Reactor Coolant System leakage (Unidentified Leakage + Identified Leakage + Reactor Coolant Pump Seal Leakage) is within the limits specified in Selected Licensee Commitment (SLC) 16.7-9, Standby Shutdown System. Corrective Action Program Report 0-C99-0606 identified that the limit specified in SLC 16.7-9 is based on the total flow capacity of the Standby Makeup Pump and does not account for the potentially higher seal injection temperatures during the postulated Standby Shutdown Facility (SSF) event. The higher seal injection temperatures will lead to an increase in seal leak-off flows. The total accumulative leakage limit specified in SLC 16.7-9 is 26 gpm. Corrective Action Program Report 0-C99-0606 recommends a lower limit of 20 gpm to allow for the increased Reactor Coolant Pump seal leakage expected during the SSF event. This procedure revision incorporates the recommended total accumulative Reactor Coolant Pump System leakage limit specified in 0-C99-0606.

Evaluation: The Standby Shutdown System (SSF) is designed to mitigate the consequences of certain postulated fire, security, and Station Blackout (SBO) events by providing capabilities to maintain Hot Standby conditions by controlling and monitoring vital systems from locations external to the main control room. The SSF provides an alternate and independent means (with respect to the control room, and within 10 minutes) to maintain Hot Standby conditions following a postulated fire or security event for one or both units for a period of 72 hours, and a postulated SBO event for a 4 hour coping duration. By design, the SSF is intended to respond to those low-probability events, which render both the control room and automatic safety systems inoperable. The SSF is not designed to mitigate a design basis event (i.e. seismic event or LOCA) and is, therefore, not nuclear safety related or seismically designed (except where interfaces with existing safety related systems are used). After a Design Basis Event (DBE), the SSF is not required to perform any function.

The Standby Makeup Pump (SMP) functions as part of the SSF to provide makeup capacity to the reactor coolant system and cooling flow to the reactor coolant pump seals. The reactor coolant pump seal leak-off flow is temperature dependent (i.e., the higher the temperature, the higher the leak-off flow. During normal operation, the reactor coolant pump seals are supplied from the Centrifugal Charging Pumps (CCP) drawing from the Volume Control Tank (VCT). During the SSF event, the SMP draws from the Spent Fuel

Pool (SFP). During the SSF event, there is no SFP cooling, so water injected into the reactor coolant pump seals will have a higher temperature than during normal operation. The SMP is capable of providing a makeup capacity of at least 26 gpm which is the makeup capacity required by SLC 16.7-9, Remedial Action (b). In considering the seal response to an increase in seal injection temperatures, this procedure change will lower the total accumulative reactor coolant system leakage limit to 20 gpm. This more conservative limit will maintain a leakage within the SLC limit during the postulated SSF event. This procedure revision does not impact any accident 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.

83 **Type:** Procedure **Unit:** 1

Title: Procedure PT/1/A/4200/01N, Revision 42A

Description: Procedure PT/1/A/4200/01N, Reactor Coolant System Boundary Valve Leak Rate Test was revised to incorporate the references and requirements of the Catawba Improved Technical Specifications and delete the references to the previously used Technical Specifications. The acceptance criteria of the test procedure was changed to the new requirements of the Improved Technical Specifications.

Evaluation: All changes associated with the change to the Improved Technical Specifications have been previously evaluated and approved by the NRC as a Technical Specification Amendment. This procedure revision incorporates these changes. There are no unreviewed safety questions associated with these changes. No further Technical Specifcation changes are required. No UFSAR changes are required.

100 **Type:** Procedure **Unit:** 1

Title: Procedure PT/1/A/4200/13H, Safety Injection System/Chemical and Volume Control System Check Valve Test", Revision 24

Description: This revision to the "Safety Injection System/Chemical and Volume Control System Check Valve Test" is being made to incorporate previously approved changes. In addition to those changes, the procedure is being rewritten 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 procedure PT/1/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 test has been performed prior to this test. This prerequisite will assure the Emergency Core Cooling System (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 "Safety Injection System/Chemical and Volume Control System Check Valve Test" procedure. 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 "Safety Injection System/Chemical and Volume Control System Check Valve Test" procedure. Steps have been added in this retype to ensure that the proper instruments are installed and removed. These instruments are consistent with those required by procedure PT/1/A/4400/01.

Finally, two motor operated valves (NI-162A and NV-312A) are required to have static and DP testing during the U1EOC11 refueling outage. Steps were added to perform this required testing. 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.

Evaluation: The purpose of PT/1/A/4200/13H, "Safety Injection System/Chemical and Volume Control System Check Valve Test" procedure, is to comply with the 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 to 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. 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. As mentioned above, 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.

Since this test is performed in "No Mode", with no fuel in the core, performance of the test will not adversely affect the probability of any accident previously evaluated in the UFSAR. Also, the ECCS portions of the above systems are not considered to be the initiator of any Chapter 15 accident.

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

240 Type: Procedure **Unit:** 0

Title: Procedure PT/1/A/4250/003C Revision 80A and PT/2/A/4250/003C Revision 59A

Description: Procedure PT/1/A/4250/003C Revision 80A "Turbine Driven Auxiliary Feedwater Pump #1 Performance Test" and PT/2/A/4250/003C Revision 59A "Turbine Driven Auxiliary Feedwater Pump #2 Performance Test" are being revised to close valves 1(2)SA-1 and 1(2)SA-4 against dynamic steam pressure during performance of the quarterly IWP procedure for the Turbine Driven Auxiliary Feedwater Pumps. In support of a pending license amendment submittal, it is necessary to demonstate that during a Steam Generator Tube Rupture (SGTR) concident with a Loss of Offsite Power (LOOP) and a single failure, it will be possible to isolate the Turbine Driven Auxiliary Feedwater System steam supply from the ruptured Steam Generator locally. For Steam Line B valve 1(2)SA-1 must be closed. For Steam Line B, valve 1(2)SA-4 must be closed. These actions must be completed prior to Reactor Coolant System depressurization to isolate the ruptured Steam Generator. The Auxiliary Feedwater System Turbine Driven Pumps are only operated in Mode 1 during performance of the quarterly IWP test. Therefore to demonstrate that the Turbine Driven Pump steam supply can be isolated against dynamic pressure conditions, it is necessary to revise the IWP procedures to perform this action.

Evaluation: The following describes the normal performance of the Auxiliary Feedwater Pump Turbine IWP Test:

Valves1(2)SA-1 and 1(2)SA-4 are locked open. The downstream steam supply valves, 1(2)SA-2 and 1(2)SA-5 are normally closed and open on a Auxiliary Feedwater Pump Turbine start signal. During performance of the IWP procedures as currently written, valve 1(2)SA-1 is unlocked, then closed. Then the Auxiliary Feedwater Pump Turbine is started in order to demonstrate that the steam supply from Steam Generator C via valve 1(2)SA-5 will fulfill its safety function.

Once Auxiliary Feedwater Pump Turbine speed is verified to be within the desired range and 1(2)CA-23, the miniflow to Upper Surge Tank (UST) dome check valve is demonstrated to stroke fully, flow is directed through the Auxiliary Feedwater Pump Turbine test line to the UST and verified to be within its acceptable range while verifying that the miniflow line isolates by proper actuation of valve 1(2)CA-20. Then the Auxiliary Feedwater Pump Turbine is tripped using the mechanical overspeed trip lever to demonstrate its proper operation.

After the pump is switched to the "off" position, and closure of valves 1(2)SA-2 and 1(2)SA-5 is verified, the mechanical overspeed trip linkage is reset, and 1(2)SA-145, the Auxiliary Feedwater Pump Turbine Stop Valve, is opened to bleed steam off the supply line and return the Auxiliary Feedwater Pump Turbine to its standby alignment.

Valve 1(2)SA-1 is then reopened and locked in its normal alignment, and valve 1(2)SA-4 is unlocked, then closed. Then the Auxiliary Feedwater Pump Turbine is started in order to demonstrate that the steam supply from Steam Generator C via valve 1(2)SA-2 will fulfill its safety function.

Once Auxiliary Feedwater Pump Turbine speed is again verified to be within the desired range, flow is again directed through the Auxiliary Feedwater Pump Turbine test line to the UST and throttled to its acceptable range. After a period of pump flow stabilization, TDH verification and vibration readings are performed to fulfill the IWP requirements. Upon completion of these readings, the Auxiliary Feedwater Pump Turbine is stopped again and realigned to its standby readiness alignment in accordance with OP/1(2)/A/6250/002, the Auxiliary Feedwater System operating procedure.

The following describes the proposed change to the performance of the Auxiliary Feedwater Pump Turbine IWP Test:

Valve 1(2)SA-4, is unlocked, then closed. The Auxiliary Feedwater Pump Turbine is started in its normal standby alignment in accordance with procedure OP/1(2)/A/6250/002. An operator is directed to unlock and close valve 1(2)SA-1 against dynamic steam pressure. Personnel performing the procedure in the Auxiliary Feedwater Pump Room are directed to verify that closing valve 1(2)SA-1 stops the Auxiliary Feedwater Pump Turbine, demonstrating that valve 1(2)SA-1 adequately isolates flow from Steam Generator B against dynamic pressure. The Control Room operator is directed to stop the Auxiliary Feedwater Pump Turbine and the operator is directed to reopen valve 1(2)SA-4. Stopping the Auxiliary Feedwater Pump Turbine first closes valves 1(2)SA-2 and 1(2)SA-5, reducing the effort necessary to reopen valve 1(2)SA-4. The Control Room Operator is then directed to restart Auxiliary Feedwater Pump Turbine. The procedure continues through its normal sequence to verify that Auxiliary Feedwater Pump Turbine speed is within the desired range and demonstrate that valve 1(2)CA-23 strokes fully. It also verifies that flow directed through the Auxiliary Feedwater Pump Turbine test line to the UST is within its acceptable range and that the miniflow line isolates by proper actuation of valve 1(2)CA-20. Then the Auxiliary Feedwater Pump Turbine is tripped using the mechanical overspeed trip lever to demonstrate its proper operation. Following reset and realignment under the normal sequence of the procedure, valve 1(2)SA-4 is left open, and the Auxiliary Feedwater Pump Turbine is restarted. The operator is directed to close valve 1(2)SA-4 against dynamic steam pressure. Personnel performing the procedure in the Auxiliary Feedwater Pump Room are directed to verify that closing valve 1(2)SA-4 stops the Auxiliary Feedwater Pump Turbine, demonstrating that valve 1(2)SA-4 adequately isolates flow from Steam Generator C against dynamic pressure. The Control Room Operator is directed to stop the Auxiliary Feedwater Pump Turbine, and the Operator is directed to reopen and re-lock valve 1(2)SA-1 and 1(2)SA-4. As before, stopping the Auxiliary Feedwater Pump Turbine closes valve 1(2)SA-2 and 1(2)SA-5, reducing the effort necessary to reopen 1(2)SA-1 and 1(2)SA-4. The procedure continues through its normal sequence to verify that Auxiliary Feedwater Pump Turbine speed is within the desired range, flow is again directed through the Auxiliary Feedwater Pump Turbine test line to the UST and throttled to its acceptable range. After a period of pump flow stabilization, TDH verification and vibration readings are performed to fulfill the IWP requirements. Upon completion of these readings, the Auxiliary Feedwater Pump Turbine is stopped again and realigned to its standby readiness alignment in accordance with the Auxiliary Feedwater system operating procedure.

Normal operation of the Auxiliary Feedwater Pump Turbine is described in UFSAR Section 10.4.9. During performance of the IWP procedure, the affected Auxiliary Feedwater Pump Turbine is declared inoperable. A 72 hour LCO is entered in accordance with Technical Specification 3.7.5, Condition B. Therefore, single failure criteria relating to system response to a design basis accident (DBA) do not apply during the performance of the test. Although the pump is taken out of service and no longer credited for responding in a DBA during the LCO, there is procedural guidance for restoring the pump to availability. Since the pump is in an LCO action statement, no reduction in the margin of safety results from this activity.

No additional components are manipulated during this procedure as revised, nor are they operated in a manner different from their normal means of operation. Only the sequence of their operation is altered. What is verified by this procedure is not so much the ability of valves 1(2)SA-1 and 1(2)SA-4 to close against dynamic steam pressure, but rather the ability of an operator to perform this action. No new failure modes or effects are postulated to result from this activity.

All systems structures and components affected by this proposed activity are involved in mitigating the consequences of previously evaluated design basis accidents. The only accident scenario associated with this activity is a postulated main steam line break, and no credible failure mode can be attributed to this activity that would increase the probability of its occurrence. Performance of this proposed activity will not result in an unreviewed safety question. No Technical Specification changes are required. No UFSAR changes are required

178 **Type:** Procedure **Unit:** 1

Title: Procedure PT/1/A/4350/12A, Diesel Generator 1A Governor and Voltage Regulator Test

Description: This procedure will replace the Unit specific procedure PT/1/A/4350/12. The procedure demonstrates acceptable response of the Diesel Generator Engine Governor and the Diesel Generator Voltage Regulator.

Evaluation: This test is identical to section 12.3 of approved procedure PT/1/A/4350/12 except that additional loads have been added to the procedure. The Containment Spray Pump, Residual Heat Removal Pump, and Fuel Pool Cooling Pump Motors have been added as additional loads. This will affect neither the probability nor consequences of accidents previously evaluated in the UFSAR. The diesel generator will be run on an isolated bus to verify acceptable performance. This test will be performed during maintenance on the diesel generator in which the governor or voltage regulator was worked on. Therefore, the 1A Diesel Generator would be considered technically inoperable. The 1B Diesel Generator will be unaffected by this test and will be able to supply emergency power if required. All pumps operated in this test will be in their normal alignments except for the Auxiliary Feedwater Pump and the Containment Spray Pump. The Containment Spray Pump will be in recirculation to the Refueling Water Storage Tank as it normally is for IWP and Heat Capacity Testing. The Auxiliary Feedwater Pump will be aligned to the Upper Surge Tank as it normally is for operability testing and for its quarterly IWP Test. No possibility of new accidents is introduced by this test. All equipment will be operated in alignments previously show to be acceptable. No unusual testing configurations are used for this test. There are no unreviewed safety questions associated with this procedure change. No Technical Specification changes are required. No UFSAR changes are required.

239 **Type:** Procedure **Unit:** 1

Title: Procedure PT/1/A/4350/12B Revision 1A, Diesel Generator 1B Govenor and Voltage Regulator Test

Description: The purpose of this procedure is to demonstrate acceptable response of the Diesel Generator governor and voltage regulator to load changes. Revision 1 makes the following changes:

1. The sequence of the test has been revised to perform the half load and full load rejection sections while paralleled to the grid before the isolated bus section. The purpose of this change is to put the governor and voltage regulator through transients prior to performing the isolated bus testing.

2. An Emergency Start signal will be simulated prior to performing the isolated bus portion to disable the nonemergency trip signal. A non-emergency trip will annunciate on the engine control panel.

Evaluation: Neither the probability nor consequences of an accident previously evaluated in the UFSAR will be increased by this test. The Diesel Generator will be run on an isolated bus to verify acceptable performance. This test will be performed following maintenance on the Diesel Generator in which the governor and/or voltage regulator were worked on or replaced; therefore, the 1B Diesel Generator will already be considered technically inoperable. The 1A Diesel Generator will be unaffected by this test and will be able to supply emergency power if required. All the pumps operated in this test will be run in their normal alignment except for the Auxiliary Feedwater Pump and Containment Spray Pump. The Containment Spray Pump will be in recirculation to the Refueling Water Storage Tank as it is at other times for IWP and Heat Capacity testing. The Auxiliary Feedwater Pump will be aligned to the Upper Surge Tank as it is for monthly operability testing and for the quarterly IWP test. The possibility of an accident different than those evaluated in the UFSAR is not created by this test. All equipment will be run in alignments that have been demonstrated to be acceptable. The probability of a malfunction of equipment important to safety will not be increased by this test. No unusual testing configurations are being used for this test. All of the equipment is being run in proven alignments. Only B train will be affected by this test and A train will be capable of meeting any accident requirements. The margin of safety as defined in the bases to Technical Specifications will not be increased by performance of this test. There are no unreviewed safety questions associated with this procedure change. No Technical Specification changes are required. No UFSAR changes are required.

194 **Type:** Procedure **Unit:** 1

Title: Procedure PT/1/A/4400/006E, Diesel Generator Engine Cooling Water System Heat Exchanger 1A Heat Capacity Test, Revision 15

Description: This revision to procedure PT/l/A/4400/006E, Diesel Generator Engine Cooling Water System Heat Exchanger 1A Heat Capacity Test, makes three major changes to the test method:

1. Provides guidance for use of a new data acquisition computer program.
2. Changes the acceptance criteria from tube side fouling factor to a overall fouling factor.
3. Provides guidance on the use of computer program Proto-HX to calculate fouling factors.

The existing data acquisition program, Dlog, is supplemented by a program based on the Labview platform (a software product from National Instruments). This program acquires data from the same points as Dlog, with the exception of the Diesel Generator Engine Cooling Water System inlet to the Lube Oil Coolers, which is not used for any calculations or trending. Instrumentation connection is unchanged, as is diesel generator operation during the test.

New acceptance criteria was calculated by calculation CNC-1223.59-01-0005 and Diesel Generator Engine Cooling Water System Test Acceptance Criteria (CNTC-1609-KD.H001 and CNTC-1609-KD.H002) were revised accordingly.

The new acceptance criteria are based on an overall fouling factor. Computer program "Proto-HX" is used to calculate the fouling factor.

Evaluation: There are no unreviewed safety questions associated with this procedure revision. The new acceptance criteria calculated in CNC-1223.59-01-0005 demonstrated that the new fouling limits will ensure that the Diesel Generator Engine Cooling Water System heat exchangers have sufficient heat transfer capacity during all postulated accidents to maintain diesel generator operability. The new acceptance criteria is based on an overall fouling factor. Previously, shell side fouling was assumed to be equal to the design fouling factor and acceptance criteria was based on tubeside fouling only. This represents a major change in the test method. The new computer program for data acquisition is a SDQA level B program and has been benchmarked to ensure it is accurately calculating process variables from voltage signals from the instrumentation. Protol-HX is a SDQA level B program that has been benchmarked to ensure it accurately calculates heat exchanger fouling factors from test data. Operation of the diesel generator is unchanged from the previously evaluated revision of this procedure, which determined that the probability of occurrence of an accident previously evaluated in the SAR would not be increased. No new accident scenarios are created. No Technical Specification chnages are required. No UFSAR changes are required.

The new acceptance criteria ensure that the Diesel Generator Engine Cooling Water System heat exchangers have sufficient heat transfer capacity during all postulated accidents to maintain the diesel generator operable.

195 **Type:** Procedure **Unit:** 1

Title: Procedure PT/1/A/4400/006F, Diesel Generator Engine Cooling Water System Heat Exchanger 1B Heat Capacity Test, Revision 17

Description: This revision to procedure PT/l/A/4400/006F, Diesel Generator Engine Cooling Water System Heat Exchanger 1B Heat Capacity Test, makes three major changes to the test method:

1. Provides guidance for use of a new data acquisition computer program.
2. Changes the acceptance criteria from tube side fouling factor to a overall fouling factor.
3. Provides guidance on the use of computer program Proto-HX to calculate fouling factors.

The existing data acquisition program, Dlog, is supplemented by a program based on the Labview platform (a software product from National Instruments). This program acquires data from the same points as Dlog, with the exception of the Diesel Generator Engine Cooling Water System inlet to the Lube Oil Coolers, which is not used for any calculations or trending. Instrumentation connection is unchanged, as is diesel generator operation during the test.

New acceptance criteria was calculated by calculation CNC-1223.59-01-0005 and Diesel Generator Engine Cooling Water System Test Acceptance Criteria (CNTC-1609-KD.H001 and CNTC-1609-KD.H002) were revised accordingly.

The new acceptance criteria are based on an overall fouling factor. Computer program "Proto-HX" is used to calculate the fouling factor.

Evaluation: There are no unreviewed safety questions associated with this procedure revision. The new acceptance criteria calculated in CNC-1223.59-01-0005 demonstrated that the new fouling limits will ensure that the Diesel Generator Engine Cooling Water System heat exchangers have sufficient heat transfer capacity during all postulated accidents to maintain diesel generator operability. The new acceptance criteria is based on an overall fouling factor. Previously, shell side fouling was assumed to be equal to the design fouling factor and acceptance criteria was based on tubeside fouling only. This represents a major change in the test method. The new computer program for data acquisition is a SDQA level B program and has been benchmarked to ensure it is accurately calculating process variables from voltage signals from the instrumentation. Proto-HX is a SDQA level B program that has been benchmarked to ensure it accurately calculates heat exchanger fouling factors from test data. Operation of the diesel generator is unchanged from the previously evaluated revision of this procedure, which determined that the probability of occurrence of an accident previously evaluated in the SAR would not be increased. No new accident scenarios are created. No Technical Specification changes are required. No UFSAR changes are required.

The new acceptance criteria ensure that the Diesel Generator Engine Cooling Water System heat exchangers have sufficient heat transfer capacity during all postulated accidents to maintain the diesel generator operable.

101 Type: Procedure **Unit:** 1

Title: Procedure PT/1/A/4400/014, Nuclear Service Water System to Auxiliary Feedwater System Suction Piping Flow Measurement, Revision 0

Description: Procedure PT/1/A/4400/014, Nuclear Service Water System to Auxiliary Feedwater System Suction Piping Flow Measurement, Revision 0, measures the flowrate through the Nuclear Service Water System to the Auxiliary Feedwater System suction supply line while the Nuclear Service Water System is in the alignment for flushing this line. Pressure readings are also taken on upstream and downstream portions of the piping in order to determine the piping pressure drop. This data is necessary to ensure that Nuclear Service Water System to Auxiliary Feedwater System suction piping can supply the required flowrate to the Auxiliary Feedwater System to maintain its operability.

Procedure PT/1/A/4400/014 was modeled after the Nuclear Service Water System to Auxiliary Feedwater System Suction Piping Flush Procedure PT/1/A/4200/059. The test procedure has adequate guidance to ensure that both the Auxiliary Feedwater System and Nuclear Service Water System are not operated outside of their design limits. During this test there will be flow established through the Nuclear Service Water System to Auxiliary Feedwater System suction line. Although this line is normally not in service during a Nuclear Service Water System flow balance, the procedure ensures that the assumptions of the Nuclear Service Water System flow balance remain valid (by requiring that the flush line be isolated within fifteen minutes of a safety injection or an automatic start of the Auxiliary Feedwater System on either Unit. The procedure contains guidance that will restore both the Nuclear Service Water System and the Auxiliary Feedwater Systems to their "Pre-test" condition if a safety injection or an automatic start of the Auxiliary Feedwater System occurs on either Unit.

Evaluation: There is no unreviewed safety question associated with this procedure. The procedure has adequate guidance to ensure that the Auxiliary Feedwater System, the Nuclear Service Water System and the Emergency Core Cooling System are not operated outside their design parameters. During this test there will be flow established through the Nuclear Service Water to Auxiliary Feedwater System suction line. Although this line is normally not in service during a Nuclear Service Water System flow balance, the procedure ensures that the assumptions of the Nuclear Service Water System flow balance remain valid. The procedure contains guidance that will restore both the Nuclear Service Water System and the Auxiliary Feedwater Systems to their "Pre-test" condition if a safety injection or an automatic start of the Auxiliary Feedwater System occurs on either Unit. No Technical Specification changes are required. No UFSAR changes are required.

102 **Type:** Procedure **Unit:** 1

Title: Procedure PT/1/A/4400/014, Nuclear Service Water System to Auxiliary Feedwater System Suction Piping Flow Measurement, Revision 1

Description: Procedure PT/1/A/4400/014 establishes flow through the Nuclear Service Water System to Auxiliary Feedwater System Suction Piping in order to determine if the piping roughness is acceptable for the Auxiliary Feedwater System to meet its design basis requirements. The procedure currently requires two operable Nuclear Service Water Pumps on the train being tested. The purpose of the requirement for maintaining two operable Nuclear Service Water Pumps is to ensure that opposite unit train related Auxiliary Feedwater System supply from the Nuclear Service Water System remains operable. This change provides an option of isolating the Auxiliary Feedwater System Turbine Driven Pump suctions from the Nuclear Service Water System with either unit in Mode 4, 5, 6 or No Mode instead of requiring two Nuclear Service Water System Pumps to be operable on the tested train. This option is only used when both units are in Mode 4, 5, 6 or No Mode. In these modes the Auxiliary Feedwater System Turbine Driven Pump is not required to be operable.

Evaluation: There are no unreviewed safety questions associated with this procedure revision. The procedure is only used when both units are in Mode 4, 5, 6 or No Mode. In these modes the Auxiliary Feedwater System Turbine Driven Pump is not required to be operable. Therefore, since the Auxiliary Feedwater System is not an accident initiator and is not required for accident mitigation in Modes 4, 5, 6 and No Mode, this change will not affect the probability or consequences of accidents analyzed in the UFSAR. This change has no effect on operability of the Nuclear Sevice Water System. No Technical Specification changes are required. No UFSAR changes are required.

179 Type: Procedure **Unit:** 0

Title: Procedure PT/1/A/4450/005A Revision 40, PT/1/A/4450/005B Revision 45, PT/2/A/4450/005A Revision 43, PT/2/A/4450/005B Revision 30

Description: Procedures PT/1/A/4450/005A Revision 40, PT/1/A/4450/005B Revision 45, PT/2/A/4450/005A Revision 43, PT/2/A/4450/005B Revision 30 were revised to correct a problem regarding verification of Engineered Safety Features (ESF) response times for the Hydrogen Skimmer Fans (HSFs) and Containment Air Return Fans (CARFs). The actual starting and timing of these fans was removed from the ESFAS Periodic Test procedures, PT/1(2)/A/4200/09, in order to prevent the introduction of foreign material into the Reactor Coolant System during refueling outages. The ESFAS procedures were changed to only verify the load sequencer contacts operated properly and take credit for the fan timing and starts with the quarterly system performance tests, PT/1/A/4450/05A(B).

After the ESFAS procedures were changed, the acceptance criteria for PT/1(2)/A/4450/05A(B) was not revised to account for the potential one second instrumentation and 11 second diesel generator startup and load sequencer group 1 delays, which total 12 seconds. The subject procedure changes will revise the maximum fan start times to account for a potential Loss of Offsite Power (LOOP) event concurrent with a design basis Loss of Cooling Accident (LOCA).

Evaluation: The design basis of the Containment Air Return Fan and Hydrogen Skimmer Fan Systems is to provide sufficient circulation of air and steam to allow the ice condenser to maintain pressures less than the containment design pressure of 15 psig, and provide sufficient mixing of hydrogen from isolated pockets and dead-ended spaces within the lower compartments to allow the Hydrogen Recombiners to reduce the concentration of hydrogen to less than 4% volume.

Both of these fans are designed to operate after a High Energy Line Break (HELB) or Loss of Coolant Accident (LOCA). These fans are designed to start within a maximum of 600 seconds time delay after a containment hi hi pressure (Sp) signal .

Technical Specification Surveillance Requirements (SR) 3.3.2, 3.6.8.4 and 3.6.11.1 verify the CARF and HSF can perform their design basis functions within 600 seconds after a design basis HELB, LOOP concurrent with a LOCA, or LOCA only. Since the ESF Response Times are no longer verified in the ESFAS Periodic Test, the acceptance criteria for the maximum fan start time must be reduced to account for the possible delay associated with a LOOP concurrent with a LOCA. The procedure acceptance criteria 11.1.1, 11.1.5, 11.3, 11.4.1, and 11.5.1 currently satisfy Technical Specification Surveillance Requirements 3.6.8.4 and 3.6.11.1. In order to satisfy SR 3.3.2 also, the fan maximum start times will be reduced by 12 seconds. The 12 seconds accounts for the Containment Spray System pressure transmitter delay (1 second in PT/1(2)/A/4200/09) and diesel generator startup and load sequencer group 1 (11 seconds in PT/1(2)/A/4200/09 and UFSAR Table 7-15).

The current procedure maximum fan start time is 599.8 seconds, where 0.2 seconds accounts for the Operator Aid Computer (OAC) scan error. Deducting 12 seconds from the current acceptance criteria yields 587.8 seconds. Therefore, the upper limit of the procedure acceptance criteria mentioned above will be reduced to 587.8 seconds to account for the ESF Response Time delays during a LOOP/LOCA.

This change will ensure the CARFs and HSFs can perform their design bases functions as described in UFSAR Section 6.2. within 600 seconds after an accident has occurred, as required by SR 3.3.2.10.

These procedure changes will not affect any of the accident initiators described in the SAR. Reducing the upper limit of the CARF and HSF start time delay acceptance criteria will not cause an inadvertent Hydrogen Skimmer System actuation. Therefore the probability of occurrence of an accident previously evaluated in the SAR will not increase.

Reducing the upper limit fan start time delay acceptance criteria will ensure that SR 3.3.2.10 is achievable, which will ensure that the assumptions used in the Catawba safety analyses are satisfied. These changes will not adversely impact operation of the CARFs and HSFs. These procedure changes will not impact any other safety related equipment. Therefore, the probability of a malfunction of equipment important to safety previously evaluated in the SAR will not increase.

These procedure changes will ensure the CARFs and HSFs perform their design basis functions as described in the SAR. No other safety related equipment necessary for accident mitigation will be affected. Therefore, the consequences of an accident previously evaluated in the SAR will not increase.

These procedure changes will ensure the CARFs and HSFs function as described in the SAR. No other safety related equipment necessary for accident mitigation will be affected. Therefore, the consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not increase.

These procedure changes will not inhibit the HSFs, CARFs, or any other safety related components from performing their design basis functions. These procedure changes will ensure the Hydrogen Skimmer System operates as designed. The assumptions used to perform the Catawba safety analyses as described in UFSAR Section 6.2 and Chapter 15 will not be affected. Therefore, the possibility of an accident of a different type than any previously evaluated in the SAR will not be created.

The CARF and HSF maximum start time delays were reduced to ensure the Hydrogen Skimmer System performs as designed during an accident. With the upper limit start time delay reduced, the CARFs and HSFs will start within the 8 to 10 minutes allowed by Technical Specification SR 3.6.8.3 and 3.6.8.4 and within the 600 second limit of SR 3.3.2.10 . No other safety related accident mitigation equipment will malfunction because of these activities. Therefore, the possibility for a different type of malfunction of equipment important to safety than any evaluated previously in the SAR will not be created.

There are no unreviewed safety questions associated with these procedure changes. No Technical Specification changes are required. No UFSAR changes are required.

137 **Type:** Procedure **Unit:** 1

Title: Procedure PT/1/B/4250/15, Condenser Cleanliness Test

Description: The purpose of the test described in this procedure is to gather data to assess the heat transfer capability of the Main Condenser. Data will be gathered from the operator aid computer and test equipment to perform Heat Exchange Institute calculations to help determine the condenser cleanliness factor of each condenser section. Pressure transmitters will be connected to the test tees of the sensing lines of 2CMPT6670, 2CMPT6680, and 2CMPT6690. These sensing lines also serve the pressure switches that enable the C-9 interlock. The procedure is designed to isolate the test tee during instrument installation without taking these pressure switches out of service. Operator Aid Computer indication of vacuum will be lost momentarily while the instruments are installed and again when they are removed.

Evaluation: There are no unreviewed safety questions associated with this procedure change. The activity described in the procedure has no effect on any of the initiating events for those accidents described in the UFSAR. Even if all three of the vacuum sensing lines were disconnected. The inleakage would be within the capability of the condenser steam air ejectors. The probability or consequences of accidents evaluated in the UFSAR will not be affected. No Technical Specification changes are required. No UFSAR changes are required.

167 **Type:** Procedure **Unit:** 2

Title: Procedure PT/2/A/4150/001I, Manual Reactor Coolant System Leakage calculation, revision 0B

Description: Procedure PT/2/A/7150/001I, Manual Reactor Coolant System Leakage Calculation, revision 0B, was prepared July 16, 1999. This procedure is used to perform a manual calculation of Reactor Coolant System leakage in the event that the Operator Aid Computer (OAC) is not available. The procedure substitutes for the function of PT/2/A/4150/001D, Reactor Coolant System Leakage Calculation.

This procedures verifies that identified and unidentified Reactor Coolant System leakage is within the limits specified in ITS 3.4.13b and 3.4.13c. The Reactor Coolant System leakage limits are less than 1 gallon per minute (gpm) unidentified and less than 10 gpm identified leakage.

Also, this procedure verifies that total accumulative Reactor Coolant System leakage (Unidentified Leakage + Identified Leakage + Reactor Coolant Pump Seal Leakage) is within the limits specified in Selected Licensee Commitment (SLC) 16.7-9, Standby Shutdown System. Corrective Action Program Report 0-C99-0606 identified that the limit specified in SLC 16.7-9 is based on the total flow capacity of the Standby Makeup Pump and does not account for the potentially higher seal injection temperatures during the postulated Standby Shutdown Facility (SSF) event. The higher seal injection temperatures will lead to an increase in seal leak-off flows. The total accumulative leakage limit specified in SLC 16.7-9 is 26 gpm. Corrective Action Program Report 0-C99-0606 recommends a lower limit of 20 gpm to allow for the increased Reactor Coolant Pump seal leakage expected during the SSF event. This procedure revision incorporates the recommended total accumulative Reactor Coolant Pump System leakage limit specified in 0-C99-0606.

Evaluation: The Standby Shutdown System (SSF) is designed to mitigate the consequences of certain postulated fire, security, and Station Blackout (SBO) events by providing capabilities to maintain Hot Standby conditions by controlling and monitoring vital systems from locations external to the main control room. The SSF provides an alternate and independent means (with respect to the control room, and within 10 minutes) to maintain Hot Standby conditions following a postulated fire or security event for one or both units for a period of 72 hours, and a postulated SBO event for a 4 hour coping duration. By design, the SSF is intended to respond to those low-probability events, which render both the control room and automatic safety systems inoperable. The SSF is not designed to mitigate a design basis event (i.e. seismic event or LOCA) and is, therefore, not nuclear safety related or seismically designed (except where interfaces with existing safety related systems are used). After a Design Basis Event (DBE), the SSF is not required to perform any function.

The Standby Makeup Pump (SMP) functions as part of the SSF to provide makeup capacity to the reactor coolant system and cooling flow to the reactor coolant pump seals. The reactor coolant pump seal leak-off flow is temperature dependent (i.e., the higher the temperature, the higher the leak-off flow. During normal operation, the reactor coolant pump seals are supplied from the Centrifugal Charging Pumps (CCP) drawing from the Volume Control Tank (VCT). During the SSF event, the SMP draws from the Spent Fuel

Pool (SFP). During the SSF event, there is no SFP cooling, so water injected into the reactor coolant pump seals will have a higher temperature than during normal operation. The SMP is capable of providing a makeup capacity of at least 26 gpm which is the makeup capacity required by SLC 16.7-9, Remedial Action (b). In considering the seal response to an increase in seal injection temperatures, this procedure change will lower the total accumulative reactor coolant system leakage limit to 20 gpm. This more conservative limit will maintain a leakage within the SLC limit during the postulated SSF event. This procedure revision does not impact any accident 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.

20 **Type:** Procedure **Unit:** 2

Title: Procedure PT/2/A/4200/009A, Revision 161B

Description: Restricted revision 161B to procedure PT/2/A/4200/009A is being made because valve 2NM-220B will not stroke to the full closed position. Valve 2NM-220B is a containment isolation valve in the Nuclear Sampling System. This revision will change the procedure to allow testing in spite of the problem with the valve. This revision will only affect enclosures 13.34 and 13.40 of the procedure. The first of these enclosures tests the containment isolation phase A, train B relay K606. In this enclosure a simulated phase A signal is given and several valves are verified to go closed. Then there is a step to verify that certain Component Cooling System and Nuclear Sampling System valves will not open using the control room controls. Valve 2NM-220B is one of these valves. In order for the valve to be in the proper position at the beginning of the test, the test line-up requires the valve to be in the closed position. Due to a problem with this valve it is impossible for the valve to go to the closed position. Therefore the valve line-up will be changed to require the valve to be in the closed or intermediate position. This will allow the test to be performed even though valve 2NM-220B is inoperable. Starting the test with the valve in the intermediate position will not significantly change the test procedure. The enclosure will still verify that the valve will not open after a phase A signal is received.

The other enclosure being modified is Enclosure 13.40. This enclosure tests the containment isolation phase A, train B relays K605 and K647. In this enclosure valve 2NM-220B is opened, a simulated phase A signal is given, and the valve is verified to go closed (step 1.7). Then the phase A signal is reset and the valve is again verified to be closed (step 1.9). This procedure revision will change the wording of steps 1.7 and 1.9 to verify that valve 2NM-220B is in the closed or intermediate position (instead of just the closed position).

The changes described above will not significantly affect the test. The purpose of the test is to verify the circuitry for the valve. The changes that are being made do not affect the ability of the test to verify the circuitry. Per Engineering, the problem with the valve is mechanical (the valve is sticking and will not completely stroke closed). The problem does not affect the control circuitry for the valve. Verifying that the valve strokes to the intermediate position is adequate to ensure that the valve circuitry is operating properly.

Evaluation: Valve 2NM220B is located on the sample line from the Steam Generator D blowdown line (inside containment). It is normally open during power operation and shutdown conditions to provide sample flow to the CT lab for continuous monitoring of Steam Generator chemistry. This valve automatically closes upon receipt of a St signal to ensure containment isolation or a Auxiliary Feedwater Motor Driven Pump B Autostart signal to prevent feedwater losses from the secondary side of the steam generators. Currently this valve is open and incapable of fully closing. Therefore, in order to ensure compliance with Technical Specification 3.6.3, valve 2NM-221A (inside containment isolation valve on the same penetration) is closed with power removed.

Procedure PT/2/A/4200/009 "Auxiliary Safeguards Periodic Test", verifies that valve 2NM-220B closes on the appropriate signals and also that the valve remains closed once a signal is received (until the signal and the valve are reset). Revision 161B to the procedure will allow testing to be performed per Enclosures 13.34 and 13.40 with valve 2NM-220B incapable of stroking to the fully closed position. This is acceptable because the problem with the valve is mechanical (binding) so that it will not go full closed. The purpose of the Auxiliary Safeguards procedure is to verify the circuitry for the valve. The procedure will continue to do this. The procedure will be changed to verify that the valve goes to the closed or intermediate position upon receipt of a safety signal instead of verifying that it goes to the closed position. This is adequate to verify that the circuitry for the valve is operating properly.

This procedure revision does not significantly alter the test procedure or adversely affect the acceptance criteria of the procedure. There are no unreviewed safety questions associated with this procedure change. No Technical Specification changes are required. No UFSAR changes are required.

140 **Type:** Procedure **Unit:** 2

Title: Procedure PT/2/A/4200/01N, Reactor Coolant System Pressure Boundary Leak Rate

Description: The purpose of Procedure PT/2/A/4200/01N is to verify that the leakage past any Reactor Coolant System Pressure Boundary Valve (PBV) does not exceed the value specified in Improved Technical Specification 3.4.14 by satisfying the 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 that bypasses containment. The valves tested are the first and second stage check valves in the Safety Injection System cold leg and hot leg injection lines as well as the Residual Heat Removal System suction isolation valves off the B and C hot legs. This revision to the procedure allows for testing of the Residual Heat Removal Suction Isolation Valves in Mode 4 and removes system vent/drain valves from the valve verification checklist. To allow for higher temperatures related to testing in Mode 4 a cooler will be added to the PBV test rig which will reduce the test fluid to approximately 130 degree F. The vent and drain valves removed from the PBV checklist were duplicate verifications which are verified by the Safety Injection System Drain, Fill, and Vent procedure.

Evaluation: There are no unreviewed safety questions associated with this procedure change. None of these procedure changes affected the test method or acceptance criteria of the test. Impact on plant operation was reviewed and procedural steps were provided to ensure no design limits were exceeded. This change will have no effect on the probability or consequences of accidents described in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required.

82 Type: Procedure **Unit:** 2

Title: Procedure PT/2/A/4200/01N, Revision 38A, Reactor Coolant System Boundary Valve Leak Rate Test

Description: Procedure PT/2/A/4200/01N, Reactor Coolant System Boundary Valve Leak Rate Test was revised to incorporate the references and requirements of The Catawba Improved Technical Specifications and delete the references to the previously used Technical Specifications. The acceptance criteria of the test procedure was changed to the new requirements of the Improved Technical Specifications.

Evaluation: All changes associated with the change to the Improved Technical Specifications have been previously evaluated and approved by the NRC as a Technical Specification Amendment. This procedure revision incorporates these changes. There are no unreviewed safety questions associated with these changes. No further Technical Specifcation changes are required. No UFSAR changes are required.

193 **Type:** Procedure **Unit:** 2

Title: Procedure PT/2/A/4200/09 Revision 140X, Engineered Safety Features Actuation Periodic Test

Description: Engineeered Safety Features (ESF) testing is frequently scheduled during Mode 5 or 6 when the Reactor Coolant System is at a level greater than 16 % Wide Range Level but still below a "Loops Filled" condition. PT/2/A/4200/09 (Engineering Safety Features Actuation Periodic Test) covers ESF Testing. For the ESF test, valve NI-173A (Residual Heat Removal System Header "A" to Reactor Coolant System Loops C and D Cold Legs Isolation Valve) or NI-178B (ND Header "B" to Reactor Coolant System Loops 'A' and 'B' Cold Legs Isolation Valve) must be closed, depending on the train of the Residual Heat Removal System being tested. This is to prevent unwanted temperature transients on the Reactor Coolant System. Since this valve must be closed for the test, the following question must be addressed: Is the "in service" Train of Residual Heat Removal still considered operable with NI-173A or NI-178B closed? Another question is whether there are other test evolution issues that could also be considered to affect train operability during the test. The Residual Heat Removal System on both units is of a similar design such that this evaluation can be used for both units.

General Design Criterion 34 requires that a system to remove residual heat be provided. The safety function of this required system is to transfer fission product decay heat and other residual heat from the core at a rate sufficient to prevent fuel or pressure boundary design limits from being exceeded. Safety grade systems are provided in the plant design, both Nuclear Steam Supply System (NSSS) scope and Balance of Plant (BOP) scope, to perform this function.

The function of the Residual Heat Removal system is to transfer heat from the Reactor Coolant System to the Component Cooling system in support of unit shutdown activities. It is required to reduce the Reactor Coolant System temperature to cold shutdown conditions at a controlled rate and maintain these conditions until the unit is started up. When the Residual Heat Removal system functions as part of Emergency Core Cooling System (ECCS), it is required to provide inventory control and cooling to the core for the injection and recirculation phases of a design basis accident.

The Residual Heat Removal System is provided with two residual heat removal pumps and heat exchangers arranged in two separate, independent flow paths. To assure reliability, each residual heat removal pump is connected to a different vital bus. Each train is isolated from the Reactor Coolant System on the suction side by two motor operated valves in series with each valve receiving power via a separate motor control center and a different vital bus. The power sources for the motor control centers are separate and redundant such that a single failure will not prevent accomplishment of the safety function of these valves which is to isolate the suction line. Each suction isolation valve is also interlocked to prevent exposure of the Residual Heat Removal System to the normal operating pressure of the Reactor Coolant System. (See UFSAR Section "Control" in topic 5.4.7.2.3.)

Residual Heat Removal System operation for normal conditions and for major failures is accomplished completely from the control room when power is restored to the suction isolation valves that have had power removed. This action is discussed in UFSAR Section "Manual Actions" in topic 5.4.7.2.7. Which states: "The Residual Heat Removal is designed to be fully operable from the control room for normal operation when power is restored to the suction isolation valves that have had power removed." This requires an operator be dispatched to the valve motor control center to close the motor control center compartment breaker. Manual operations required of the operator are: restoring power to the suction isolation valves that have power removed, opening the suction isolation valves, positioning the flow control valves down stream of the Residual Heat Removal Heat Exchangers, and starting the Residual Heat Removal Pumps.

Although valves NI-173A and NI-178B are not stated as specific examples of manual actions, the same philosophy would still apply. Either of these valves may be aligned manually by simply resetting the Diesel Generator (D/G) Load Sequencer, aligning the valve, and placing the test train of Residual Heat Removal System in service in the Residual Heat Removal mode.

The redundancy in the Residual Heat Removal System design provides the system with the capability to maintain its cooling function even with major single failures, such as failure of an Residual Heat Removal pump, valve, or heat exchanger without impact on the redundant train's continued heat removal.

One example of a failure of this type refered to in the UFSAR is a failure in the interlock circuitry which is designed to prevent exposure of the Residual Heat Removal System to the normal operating pressure of the Reactor Coolant System (See UFSAR Section "Control" in topic 5.4.7.2.3). In the event of such a failure, Residual Heat Removal System operation can be initiated by defeating the failed interlock through corrective action at the Solid State Protection System cabinet or at the individual affected motor control centers. Again this is an example that even though valves NI-173A and NI-178B are not stated as specific examples of manual actions, the same philosophy would still apply. Either of these valves may be aligned manually by resetting the Diesel Generator (D/G) Load Sequencer, aligning the valve, and placing the test train of Residual Heat Removal in service in the residual heat removal mode. The manual action to open either valve NI-173A or NI-178B is considered to be a relatively easy task compared to the task of actually putting the redundant train of Residual Heat Removal in service in the residual heat removal mode.

The Residual Heat Removal System is designed to be fully operable from the control room when power is restored to the suction isolation valves that have had power removed. Manual operations required of the operator are: opening the suction isolation valves, positioning the flow control valves downstream of the Residual Heat Removal System heat exchangers, and starting the Residual Heat Removal pumps. By nature of its redundant two train design, the Residual Heat Removal System is designed to accept all major component single failures with the only effect being an extension in the required cooldown time. For two low probability electrical system single failures, i.e., failure in the suction isolation valve interlock circuitry, or diesel generator failure in conjunction with loss of offsite power, limited operator action outside the control room is required to open the suction isolation valves. Manual actions are discussed in more detail in Sections "Manual Actions" in topic 5.4.7.2.7. (This is in addition to the normal procedure for restoring power to the suction isolation valves that have had power removed.) The only motor operated valves in the Residual Heat Removal System which are subject to flooding are the suction isolation valves which are not required to function after a loss of coolant accident. Although Westinghouse considers it to be of low probability, spurious operation of a single motor operated valve can be accepted without loss of function as a result of the redundant two-train design.

The other type of failure given as an example in the UFSAR, which can prevent opening the Residual Heat Removal suction isolation valves from the control room, is a failure of an electrical power train. Such a failure is extremely unlikely to occur during the few minutes per year of operation time during which it can have any consequence. If such an unlikely event should occur, several alternatives are available. The most realistic approach would be to obtain restoration of offsite power, which can be expected to occur in less than half an hour. Other alternatives are to restore the emergency diesel generator to operation, to bring in an alternate power source, or to open the affected valves with their manual handwheels. During ESF Testing, one train of Residual Heat Removal is operable and in Residual Heat Removal mode on the Reactor Coolant System loops per Technical Specifications 3.4.8 or 3.9.5. This train is required to have an operable D/G to support loss of power to an essential bus. The train of Residual Heat Removal that is being tested must also be operable per Technical Specifications 3.4.8 or 3.9.5. Technical Specifications 3.4.8 or 3.9.5 does not have a requirement that the redundant train of Residual Heat Removal must be aligned in Residual Heat Removal mode. The redundant train of Residual Heat Removal must also have a functional D/G due to the nature of the test although Technical Specification 3.8.2 only requires that a train of offsite power be available. Therefore if NI-173A or NI-178B were closed for ESF Testing, the means to open the valve to align the redundant train of Residual Heat Removal in Residual Heat Removal mode is readily available.

As a result of several "Loss of Residual Heat Removal " events throughout the Nuclear Power Industry, Generic Letter 87-12, "Loss of Residual Heat Removal while the Reactor Coolant System is Partially Filled", was issued on July 9, 1987. In response to this Generic Letter and Generic Letter 88-17, "Loss of Decay Heat Removal" (issued October 17, 1988), Catawba undertook an extensive review of the physical plant configuration, training programs for plant personnel, administrative procedures, and programmatic enhancements for the plant. The emphasis of these reviews was on improvements to plant operations while the reactor coolant system is partially filled.

The following related facts are present when performing ESF Testing at a "loops not filled" condition in mode 5 or 6: During the ESF test, the suction of Residual Heat Removal Train in test is aligned up to the loops. During the ESF test, the Residual Heat Removal Train not being tested will be in operation in Residual Heat Removal mode. Time to boiling at 16 % Reactor Coolant System level is approximately 45 minutes. This time assumes low decay heat conditions (i.e. The core has been reloaded such that 1/3 of the core is new fuel), no makeup flow initiated, and approximately eighteen days since shutdown. These values are typical for twenty-eight to thirty day refueling outages. The time to boiling will typically be longer in longer outages. Time to core uncovery at 16 % NCS level is approximately 15 hours. Required makeup flow to remove decay heat is approximately 25 gpm. This required makeup flow assumes low decay heat conditions and approximately eighteen days since shutdown. These values are typical for twenty-eight to thirty day refueling outages. The required makeup flow will typically be less in longer outages. Valve ND-32A (Residual Heat Removal Train "A" Hot Leg Injection Isolation) or ND-65B (Residual Heat Removal Train "B" Hot Leg Injection Isolation) can be used as alternate flow paths. In modes 5 or 6, there is no requirement that Residual Heat Removal must flow to all four cold legs simultaneously as in modes 1-4. In modes 1-4, flow to all four cold legs is required for core cooling during the injection mode. This accounts for the flow being lost in one leg due to a postulated break. This is not a requirement for operating in Residual Heat Removal mode. In Residual Heat Removal mode, one Residual Heat Removal pump may be used to remove decay heat by returning the water to the Reactor Coolant System through only two cold legs. Therefore by utilizing ND-32A and ND-65B one train of Residual Heat Removal can be used to inject through either NI-173A or NI-178B. The Unit Configuration is controlled by Site Directive 3.1.30 (Unit Shutdown Configuration Control) which requires two Operable Trains of Residual Heat Removal, one Operable D/G, one Operable, one Available Off-site Power Supply, one Operable Centrifugal Charging Train, one available Safety Injection Train, two Available Containment Sump Recirculation Trains. Technical Specification 3.9.5 (Mode 6 and < 23 feet) requires two operable Residual Heat Removal trains, with one in operation. Technical Specification Surveillance Requirement 3.9.5.2 does not specify that the redundant train of Residual Heat Removal be in the residual heat removal alignment, only that it has the correct breaker alignment and indicated power are available to the pump. The bases for this Technical Specification discusses that an Residual Heat Removal Pump is operable if it can receive power and be able to provide flow if required. Tech Spec 3.4.8 (Mode 5 loops not filled) requires two operable Residual Heat Removal trains, with one in operation. Technical Specification Surveillance Requirement 3.4.8.2 does not specify that the redundant train of Residual Heat Removal be in residual heat removal alignment, only that it has the correct breaker alignment and indicated power are available to the pump. However this spec does allow one Residual Heat Removal Train to be inoperable for <= 2 hours for testing surveillance providing that the other Residual Heat Removal Train is Operable. The bases for this Technical Specification discusses that a Residual Heat Removal Pump is operable if it can receive power and be able to provide flow if required. There are two possible postulated failure modes:

1. Failure of NI-173A or NI-178B to open.
2. Failure of the sequencer to actuate.

There would have to be two failures in order for a loss of Decay Heat Removal situation to arise: Failure of the "in service" Residual Heat Removal Train to remove decay heat in the residual heat removal mode. One of the two failures listed above. The UFSAR addresses manual valve alignments as being permissible once the unit is below Mode 4. However the manual alignment of NI-173A or NI-178B during testing is not specifically discussed. What is discussed is that a Residual Heat Removal Pump is operable if it can receive power and be able to provide flow if required. During the ESF test, there would not be any troubleshooting needed to place the "test" Train of Residual Heat Removal in service, it would be specifically known what the required steps were based on specific failures identified here. During the ESF Test, the 60-second timer on the sequencer is by-passed, thus allowing the operators to regain control at any time during the test. This is an action that has to be done to start the Residual Heat Removal Pump if running the Blackout portion of the ESF Test regardless of whether there is a failure of either NI-173A or NI-178B. There are no jumpers installed during the ESF Test that would interfere with Operation's ability to place the train of Residual Heat Removal being testing into Residual Heat Removal service in an expeditious manner.

The Safety Injection System Design Basis Document states that with NI-173A or NI-I78B closed and incapable of opening, the associated Residual Heat Removal Train is inoperable. During the test, NI-173A or NI- 178B would be capable of being opened. The Residual Heat Removal train being tested is still capable of being aligned for Residual Heat Removal mode. In the event of a failure of the D/G Load Sequencer, the operator may reset it at anytime to regain control of the equipment. The needed loads may then be manually aligned to the bus and the Residual Heat Removal train in test placed in service. In the event of a failure of NI-173A or NI-178B to open electrically, an operator may be dispatched into the Auxiliary Building to locally open the valve. An alternate alignment may be made locally to use valves ND-32A and ND-65B to align one Residual Heat Removal train discharge through the other train's cold leg injection isolation valve. The local opening is an option that has always been available to the operators.

The Residual Heat Removal Train being tested is considered operable during the test based on the following reasons: Actions to recover "test" Train of Residual Heat Removal will be specifically spelled out in a Pre-job Briefing.
a. How to take control of the sequencer and manually load the needed loads in the event of a sequencer failure.
b. Local operation of NI- I 73A or NI- I 78B is allowed. NI-173A and NI-178B is located in the Auxiliary Building in the Mechanical Penetration Room, thus a containment entry is not required to locally operate the valve.
c. ND-32A (Residual Heat Removal Train "A" Hot Leg Injection Isolation) or ND-65B (Residual Heat Removal Train "B" Hot Leg Injection Isolation) can be used as alternate flow paths.
d. The operators also use AP/2/A/5500/19 (Loss of Residual Heat Removal System) to recover Residual Heat Removal if there is a total loss of Residual Heat Removal. This procedure walks the operators though all the steps required for troubleshooting a Residual Heat Removal problem and recovering that function.
NI-173A or NI-178B would be capable of being opened unless a failure occurs in the train related ESF Test. The Residual Heat Removal train being tested is still capable of being aligned for Residual Heat Removal mode. In the event of a failure of the D/G Load Sequencer, the operator may reset it at anytime to regain control of the equipment. The needed loads may then be manually aligned to the bus and the Residual Heat Removal train in test placed in service. In the event of a failure of NI-173A or NI-178B to open electrically, an operator may be dispatched into the Auxiliary Building to locally open the valve. An alternate alignment may be made locally to use ND-32A and ND-65B to align one Residual Heat Removal train discharge through the other train's cold leg injection isolation valve. The local opening is an option that has always been available to the operators. There is more than adequate Thermal Margin for Operations to recover the Residual Heat Removal Train being tested if required. The unit is in Low Decay Heat Conditions, which allow much more Thermal Margin than High Decay Heat Conditions. There would have to be two failures in order for this situation to arise. Reactor Coolant System level is maintained at > 16 %, even though the system is not in a "Loops Filled" condition. This Level minimizes the chances that venting of the suction of the Residual Heat Removal pumps would be necessary since a loss of suction is not likely. Technical Specification Surveillance Requirement 3.9.5.2 and 3.4.5.2 do not specify that the redundant train of Residual Heat Removal be in Residual Heat Removal alignment, only that it has the correct breaker alignment and indicated power are available to the pump.

The bases for this Technical Specification discusses that an Residual Heat Removal Pump is operable if it can receive power and be able to provide flow if required. Manual alignments are allowed in mode 4. During the ESF Test, the 60-second timer on the sequencer is by-passed, thus allowing the operators to regain control at any time 'during the test. During the ESF test, there would not be any troubleshooting needed to place the train of Residual Heat Removal being tested in service, it would be specifically known what the required steps were.

Evaluation: The purpose of the Residual Heat Removal System is to remove decay heat and for boron mixing in mode 5 and 6. The Residual Heat Removal System provides the decay heat removal function by cooling water removed from the Reactor Coolant System and returning it at a cooler temperature. Boron mixing is accomplished by the flow of removing water from the Reactor Coolant System hot leg and returning it to Reactor Coolant System cold legs, Redundancy is maintained by requiring two operable trains of Residual Heat Removal while in the "Loops not filled" condition. The requirement of one train being in operation on the Reactor Coolant System loops is maintained. This procedure change will not change the requirements for this plant configuration since the Residual Heat Removal train being tested can be realigned to Reactor Coolant System loops at any time during the test. Thus there is no increase in the probability of occurrence of an accident previously evaluated in the SAR. ESF Testing is designed to test all the Engineering Safeguards function built into the plant. This test would thus require that the Residual Heat Removal train being tested be functional. This same test also requires that a D/G be function since it is required to mitigate certain design basis accidents. Therefore, an increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR is not present since all the equipment on one train normally used to mitigate such design basis accidents is fully tested in this procedure. Furthermore, the plant will not be put in a jeopardized position because all the Technical Specifications Limiting Conditions for Operation and surveillance requirements will be met. No system used to mitigate any accident is degraded. One train of engineered safeguards equipment is being tested by this procedure. This means that all the equipment on that train must function in order to test it. Neither any fission product barrier or source term is affected by this change. Therefore all the equipment needed to mitigate the consequences of an accident previously evaluated in the SAR are available. The Residual Heat Removal train being tested is still capable of being aligned for Residual Heat Removal mode. In the event of a failure of the D/G Load Sequencer, the operator may reset it at anytime to regain control of the equipment. The needed loads may then be manually aligned to the bus and the Residual Heat Removal train in test placed in service. In the event of a failure of valve NI-173A or NI-178B to open electrically, an operator may be dispatched into the Auxiliary Building to locally open the valve. An alternate alignment may be made locally to use ND-32A and ND-65B to align one Residual Heat Removal train discharge through the other train's cold leg injection isolation valve. The local opening is an option that has always been available to the operators. Thus there is no increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR. There are no new accidents introduced by performing ESF Testing during mode 5 or 6 in a "Loops not filled" condition with low decay heat conditions. There is ample Thermal Margin available for the operators to take action if needed to place the Residual Heat Removal Train being tested in service in Residual Heat Removal mode. The steps needed to quickly restore the functions of the Residual Heat Removal train being tested will be listed in the Contingency Plans of the Level 2 Pre-job Brief Tailgate Package for Operations. The D/G will still be available for electrical power if needed for alignment. The test does not perform any alignments that could not be quickly restored to normal service alignments for these modes. This activity does not create the possibility for an accident of a different type than any evaluated previously in the SAR. This test does not align ESF equipment such that it could not be relied upon to mitigate design base accidents when called upon. The procedure does not put the second required operable Residual Heat Removal train in a jeopardized alignment. There is power maintained available to NI-173A and NI-178B with a D/G to back it up. In the event of a failure of the D/G Load Sequencer, the operator may reset the sequencer and manually align any needed loads. There are handwheels on the valves that may also be used if needed. The possibility for a different type of malfunction of equipment important to safety in the SAR is not present. The level of safety required by Technical Specifications is still maintained because nothing is done to disable the ability of the operator to align the Residual Heat Removal train being tested in Residual Heat Removal mode. This allows the Residual Heat Removal System to meet its safety function of mitigation of the accidental boron dilution event by providing flow for heat removal and adequate mixing. By being able to maintain the Reactor Coolant System below 200 degrees F, this prevents the dilution of the Reactor Coolant System due to the loss of boron plating out on components near the area of boiling activity. One train of Residual Heat Removal is in operation on the Reactor Coolant System. The redundant train of Residual Heat Removal has power is available to the pump as required. The ability to align the redundant train in Residual Heat Removal mode prior to the stated 200 degree F is assured by the availability of the equipment, power sources, and operator action. There are no changes to any design limits or setpoints. No control, instrument functions, or performance of any structure, system, or component is degraded.

This procedure change does not involve an unreviewed safety question. No Technical Specifications changes are required. The train of Residual Heat Removal being tested is capable being aligned in service in Residual Heat Removal mode if needed by the operator. The closing of NI-173A or NI-178B does not change this. During the test, all the electrical requirements to open the valves at any time are maintained. There are no electrical interlocks that must be modified to open the valves. Provisions are in the Contingency Plans of the Pre-job Briefing to allow the operator to regain control of equipment being loaded on by the D/G Load Sequencer at anytime. Finally, both valves have installed handwheels which allow an operator to locally manipulate the valves in the Auxiliary Building. No UFSAR Changes are required.

22 Type: Procedure **Unit:** 2

Title: Procedure PT/2/A/4200/31A, Revision 12C

Description: Procedure PT/2/A/4200/31A ensures that the Steam Generator Power Operated Relief Valves (PORVs) can be opened in "Manual" with the design steam pressure on the valve. This revision to the procedure adds information so that it is no longer necessary to measure the differential pressure across the valve actuator. In the current procedure a differential pressure transmitter is installed across the valve actuator. The valve is then partially stroked (30% open) and the actuator differential pressure is recorded with a Visicorder. This pressure is then compared against a calculated pressure (based on actual steam pressure being less than design pressure) to determine acceptable valve performance.

The revised procedure will provide an option to have Maintenance adjust the nitrogen regulator to the calculated pressure (the method of calculating this pressure will be the same as in the current procedure). Operations will then stroke the valve 30% open. The only acceptance criteria for this test will be that the valve stem travels more than 0.75 inches. This acceptance criteria is currently in the procedure along with another criteria for verifying that the calculated actuator differential pressure is greater than or equal to the measured differential pressure. Adjusting the regulator pressure to the calculated value prior to stroking the valve accomplishes the same thing as measuring the differential pressure and comparing it against the calculated value. Either case allows verification of acceptable valve performance.

Evaluation: This procedure revision allows another option for verifying proper operation of the PORV. The revision does not allow operation of the PORV in a manner other than is specified in the current procedure. The procedure requires that the valve be declared inoperable if the nitrogen regulator pressure is adjusted and it includes steps to return the regulator to the proper setting before declaring the valve operable. There are no Unreviewed Safety Questions associated with this procedure revision. No Technical Specification changes are required. No UFSAR changes are required.

252 Type: Procedure **Unit:** 2

Title: Procedure PT/2/A/4400/006C, Component Cooling Heat Exchanger 2A Heat Capacity Test, Revision 7

Description: This procedure revision makes the following changes.
1. It provides guidance for use of a new data acquisition computer program.
2. It incorporates previously approved changes.
The existing data acquisition program, Dlog, is superceded by a program based on the Labview platform (a software product from National Instruments). This program acquires data from the same points as Dlog. Also, the program calculates heat exchanger fouling factors while the data is being collected. Therefore, the CORN003 program does not need to be used to calculate fouling factors after the data is acquired.

Evaluation: The new computer program was benchmarked to ensure that it accurately calculates process variables from voltage signals provided by monitoring instrumentation. There are no actual changes to the operation of the systems involved. There are no unreviewed safety questions associated with this procedure revision. No Technical Specification changes are required. No UFSAR changes are required.

238 Type: Procedure **Unit:** 2

Title: Procedure PT/2/A/4400/006D, Component Cooling Heat Exchanger 2B Heat Capacity Test, Revision 6

Description: This procedure revision makes the following changes.
1. It provides guidance for use of a new data acquisition computer program.
2. It incorporates previously approved changes.
The existing data acquisition program, Dlog, is superceded by a program based on the Labview platform (a software product from National Instruments). This program acquires data from the same points as Dlog. Also, the program calculates heat exchanger fouling factors while the data is being collected. Therefore, the CORN003 program does not need to be used to calculate fouling factors after the data is acquired.

Evaluation: The new computer program was benchmarked to ensure that it accurately calculates process variables from voltage signals provided by monitoring instrumentation. There are no actual changes to the operation of the systems involved. There are no unreviewed safety questions associated with this procedure revision. No Technical Specification changes are required. No UFSAR changes are required.

196 **Type:** Procedure **Unit:** 2

Title: Procedure PT/2/A/4400/006E, Diesel Generator Engine Cooling Water System Heat Exchanger 2A Heat Capacity Test, Revision 12

Description: This revision to procedure PT/2/A/4400/006E, Diesel Generator Engine Cooling Water System Heat Exchanger 2A Heat Capacity Test, makes three major changes to the test method:

1. Provides guidance for use of a new data acquisition computer program.
2. Changes the acceptance criteria from tube side fouling factor to a overall fouling factor.
3. Provides guidance on the use of computer program Proto-HX to calculate fouling factors.

The existing data acquisition program, Dlog, is supplemented by a program based on the Labview platform (a software product from National Instruments). This program acquires data from the same points as Dlog, with the exception of the Diesel Generator Engine Cooling Water System inlet to the Lube Oil Coolers, which is not used for any calculations or trending. Instrumentation connection is unchanged, as is diesel generator operation during the test.

New acceptance criteria was calculated by calculation CNC-1223.59-01-0005 and Diesel Generator Engine Cooling Water System Test Acceptance Criteria (CNTC-1609-KD.H001 and CNTC-1609-KD.H002) were revised accordingly.

The new acceptance criteria are based on an overall fouling factor. Computer program "Proto-HX" is used to calculate the fouling factor.

Evaluation: There are no unreviewed safety questions associated with this procedure revision. The new acceptance criteria calculated in CNC-1223.59-01-0005 demonstrated that the new fouling limits will ensure that the Diesel Generator Engine Cooling Water System heat exchangers have sufficient heat transfer capacity during all postulated accidents to maintain diesel generator operability. The new acceptance criteria is based on an overall fouling factor. Previously, shell side fouling was assumed to be equal to the design fouling factor and acceptance criteria was based on tubeside fouling only. This represents a major change in the test method. The new computer program for data acquisition is a SDQA level B program and has been benchmarked to ensure it is accurately calculating process variables from voltage signals from the instrumentation. Proto-HX is a SDQA level B program that has been benchmarked to ensure it accurately calculates heat exchanger fouling factors from test data. Operation of the diesel generator is unchanged from the previously evaluated revision of this procedure, which determined that the probability of occurrence of an accident previously evaluated in the SAR would not be increased. No new accident scenarios are created. No Technical Specification chnages are required. No UFSAR changes are required.

The new acceptance criteria ensure that the Diesel Generator Engine Cooling Water System heat exchangers have sufficient heat transfer capacity during all postulated accidents to maintain the diesel generator operable.

197 Type: Procedure **Unit:** 2

Title: Procedure PT/2/A/4400/006F, Diesel Generator Engine Cooling Water System Heat Exchanger 2B Heat Capacity Test, Revision 13

Description: This revision to procedure PT/2/A/4400/006F, Diesel Generator Engine Cooling Water System Heat Exchanger 2B Heat Capacity Test, makes three major changes to the test method:

1. Provides guidance for use of a new data acquisition computer program.
2. Changes the acceptance criteria from tube side fouling factor to a overall fouling factor.
3. Provides guidance on the use of computer program Proto-HX to calculate fouling factors.

The existing data acquisition program, Dlog, is supplemented by a program based on the Labview platform (a software product from National Instruments). This program acquires data from the same points as Dlog, with the exception of the Diesel Generator Engine Cooling Water System inlet to the Lube Oil Coolers, which is not used for any calculations or trending. Instrumentation connection is unchanged, as is diesel generator operation during the test.

New acceptance criteria was calculated by calculation CNC-1223.59-01-0005 and Diesel Generator Engine Cooling Water System Test Acceptance Criteria (CNTC-1609-KD.H001 and CNTC-1609-KD.H002) were revised accordingly.

The new acceptance criteria are based on an overall fouling factor. Computer program "Proto-HX" is used to calculate the fouling factor.

Evaluation: There are no unreviewed safety questions associated with this procedure revision. The new acceptance criteria calculated in CNC-1223.59-01-0005 demonstrated that the new fouling limits will ensure that the Diesel Generator Engine Cooling Water System heat exchangers have sufficient heat transfer capacity during all postulated accidents to maintain diesel generator operability. The new acceptance criteria is based on an overall fouling factor. Previously, shell side fouling was assumed to be equal to the design fouling factor and acceptance criteria was based on tubeside fouling only. This represents a major change in the test method. The new computer program for data acquisition is a SDQA level B program and has been benchmarked to ensure it is accurately calculating process variables from voltage signals from the instrumentation. Proto-HX is a SDQA level B program that has been benchmarked to ensure it accurately calculates heat exchanger fouling factors from test data. Operation of the diesel generator is unchanged from the previously evaluated revision of this procedure, which determined that the probability of occurrence of an accident previously evaluated in the SAR would not be increased. No new accident scenarios are created. No Technical Specification chnages are required. No UFSAR changes are required.

The new acceptance criteria ensure that the Diesel Generator Engine Cooling Water System heat exchangers have sufficient heat transfer capacity during all postulated accidents to maintain the diesel generator operable.

197 Type: Procedure **Unit:** 2

Title: Procedure PT/2/A/4400/006F, Diesel Generator Engine Cooling Water System Heat Exchanger 2B Heat Capacity Test, Revision 13

Description: This revision to procedure PT/2/A/4400/006F, Diesel Generator Engine Cooling Water System Heat Exchanger 2B Heat Capacity Test, makes three major changes to the test method:

1. Provides guidance for use of a new data acquisition computer program.
2. Changes the acceptance criteria from tube side fouling factor to a overall fouling factor.
3. Provides guidance on the use of computer program Proto-HX to calculate fouling factors.

The existing data acquisition program, Dlog, is supplemented by a program based on the Labview platform (a software product from National Instruments). This program acquires data from the same points as Dlog, with the exception of the Diesel Generator Engine Cooling Water System inlet to the Lube Oil Coolers, which is not used for any calculations or trending. Instrumentation connection is unchanged, as is diesel generator operation during the test.

New acceptance criteria was calculated by calculation CNC-1223.59-01-0005 and Diesel Generator Engine Cooling Water System Test Acceptance Criteria (CNTC-1609-KD.H001 and CNTC-1609-KD.H002) were revised accordingly.

The new acceptance criteria are based on an overall fouling factor. Computer program "Proto-HX" is used to calculate the fouling factor.

Evaluation: There are no unreviewed safety questions associated with this procedure revision. The new acceptance criteria calculated in CNC-1223.59-01-0005 demonstrated that the new fouling limits will ensure that the Diesel Generator Engine Cooling Water System heat exchangers have sufficient heat transfer capacity during all postulated accidents to maintain diesel generator operability. The new acceptance criteria is based on an overall fouling factor. Previously, shell side fouling was assumed to be equal to the design fouling factor and acceptance criteria was based on tubeside fouling only. This represents a major change in the test method. The new computer program for data acquisition is a SDQA level B program and has been benchmarked to ensure it is accurately calculating process variables from voltage signals from the instrumentation. Proto-HX is a SDQA level B program that has been benchmarked to ensure it accurately calculates heat exchanger fouling factors from test data. Operation of the diesel generator is unchanged from the previously evaluated revision of this procedure, which determined that the probability of occurrence of an accident previously evaluated in the SAR would not be increased. No new accident scenarios are created. No Technical Specification chnages are required. No UFSAR changes are required.

The new acceptance criteria ensure that the Diesel Generator Engine Cooling Water System heat exchangers have sufficient heat transfer capacity during all postulated accidents to maintain the diesel generator operable.

103 **Type:** Procedure **Unit:** 2

Title: Procedure PT/2/A/4400/014, Nuclear Service Water System to Auxiliary Feedwater System Suction Piping Flow Measurement, Revision 0

Description: Procedure PT/2/A/4400/014, Nuclear Service Water System to Auxiliary Feedwater System Suction Piping Flow Measurement, Revision 0, measures the flowrate through the Nuclear Service Water System to the Auxiliary Feedwater System suction supply line while the Nuclear Service Water System is in the alignment for flushing this line. Pressure readings are also taken on upstream and downstream portions of the piping in order to determine the piping pressure drop. This data is necessary to ensure that Nuclear Service Water System to Auxiliary Feedwater System suction piping can supply the required flowrate to the Auxiliary Feedwater System to maintain its operability.

Procedure P/1/A/4400/014 was modeled after the Nuclear Service Water System to Auxiliary Feedwater System Suction Piping Flush procedure PT/1/A/4200/059. The test procedure has adequate guidance to ensure that both the Auxiliary Feedwater System and Nuclear Service Water System are not operated outside of their design limits. During this test there will be flow established through the Nuclear Service Water System to Auxiliary Feedwater System suction line. Although this line is normally not in service during a Nuclear Service Water System flow balance, the procedure ensures that the assumptions of the Nuclear Service Water System flow balance remain valid (by requiring that the flush line be isolated within fifteen minutes of a safety injection or an automatic start of the Auxiliary Feedwater System on either Unit). The procedure contains guidance that will restore both the Nuclear Service Water System and the Auxiliary Feedwater Systems to their "Pre-test" condition if a safety injection or an automatic start of the Auxiliary Feedwater System occurs on either Unit.

Evaluation: There is no unreviewed safety question associated with this procedure. The procedure has adequate guidance to ensure that the Auxiliary Feedwater System, the Nuclear Service Water System and the Emergency Core Cooling System are not operated outside their design parameters. During this test there will be flow established through the Nuclear Service Water to Auxiliary Feedwater System suction line. Although this line is normally not in service during a Nuclear Service Water System flow balance, the procedure ensures that the assumptions of the Nuclear Service Water System flow balance remain valid. The procedure contains guidance that will restore both the Nuclear Service Water System and the Auxiliary Feedwater Systems to their "Pre-test" condition if a safety injection or an automatic start of the Auxiliary Feedwater System occurs on either Unit. No Technical Specification changes are required. No UFSAR changes are required.

138 **Type:** Procedure **Unit:** 2

Title: Procedure PT/2/B/4250/15, Condenser Cleanliness Test

Description: The purpose of the test described in this procedure is to gather data to assess the heat transfer capability of the Main Condenser. Data will be gathered from the operator aid computer and test equipment to perform Heat Exchange Institute calculations to help determine the condenser cleanliness factor of each condenser section.
Pressure transmitters will be connected to the test tees of the sensing lines of 2CMPT6670, 2CMPT6680, and 2CMPT6690. These sensing lines also serve the pressure switches that enable the C-9 interlock. The procedure is designed to to isolate the test tee during instrument installation without taking these pressure switches out of service. Operator Aid Computer indication of vacuum will be lost momentarily while the instruments are installed and again when they are removed.

Evaluation: There are no unreviewed safety questions associated with this procedure change. The activity described in the procedure has no effect on any of the initiating events for those accidents described in the UFSAR. Even if all three of the vacuum sensing lines were disconnected. The inleakage would be within the capability of the condenser steam air ejectors. The probability or consequences of accidents evaluated in the UFSAR will not be affected. No Technical Specification changes are required. No UFSAR changes are required.

98 Type: Procedure **Unit:** 0

Title: Procedure TI/0/A/3230/011 Revision 0

Description: Procedure TI/0/A/3230/011 Revision 0 "Procedure for Core Exit Thermocouple Connector Replacement" is for the Incore Instrumentation System, Core Exit Thermocouples. This system is used to obtain Core Exit Temperature readings from the reactor during plant operations in Mode 1 through Mode 5. Operations and Nuclear Engineering use these readings to monitor core parameters.

Asea Brown Boveri Combustion Engineering, the original equipment manufacturer, issued a new procedure to replace damaged Core Exit Thermocouple connectors. Their procedure number is 00000-SIS-057 Rev #9 ("Procedure for Core Exit Thermocouple Connector Replacement"). Modification CE-10088 adds this procedure as design document CNM 1354.12-0020 002.

The vendor procedure is being modified to add Catawba Nuclear Station specific requirements such as Prerequisites, Operations Work Notifications, Quality Control Inspections, etc. This modified procedure will be TI/O/A/3230/011 Revision 0.

Evaluation: The Core Exit Thermocouple sub-system consists of 40 nuclear safety related and 25 non nuclear-safety related instrument channels. However, all 65 thermocouples are qualified as nuclear safety related as they exit the reactor head at one of the five instrument ports. Each thermocouple is in a 5/16-inch stainless steel guide tube, which is a Reactor Coolant System pressure boundary. This procedure will allow the guide tube to be cut if needed for replacement activities. This system performs no control function, but provides Post Accident Monitoring System (PAMS) Instrumentation, and input to the Inadequate Core Cooling Monitor system and Operator Aid Computer. Both of these systems provide Core Exit Temperature Sub-Cooling data and alarms to the control room operator.

The Core Exit Thermocouple sub-system does not perform any control functions, but is a passive monitoring system, used to obtain core exit temperatures. The thermocouple guide tubes are Reactor Coolant System pressure boundaries that could initiate a small break LOCA or a small Reactor Coolant System leak. However, this procedure will restore the pressure boundary. This event would be bounded by UFSAR 15.6.5. The instructions on cutting and installing a new pressure fitting on the guide tube will ensure that the reactor coolant pressure boundary integrity is maintained. Therefore, this change will not increase the probability or consequences of an accident evaluated in the UFSAR nor are any new accident scenarios created.

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

171 **Type:** Procedure **Unit:** 0

Title: Procedure TT/0/A/9100/071 Revision 1, Temporary Test for Auxiliary Building Single Train Air Flow

Description: Procedure TT/0/A/9100/071 Revision 1, Temporary Test for Auxiliary Building Single Train Air Flow will test the Auxiliary Building Ventilation System filtered exhaust airflow. This will provide data showing the difference between single and dual train alignment airflow rates. The Auxiliary Building Ventilation System is currently tested with filter units on both trains operating in parallel. A concern was identified with high airflow if the system was operated in a single train alignment. Administrative restrictions were placed on the technical specification allowable limits for maximum airflow and carbon iodine penetration. The Auxiliary Building Ventilation System was subsequently declared operable but degraded and the previously used administrative limits were formalized. The limits currently imposed on the system are the reduction of the maximum allowable flow rates from 33,000 cfm to 32,000 cfm and the carbon iodine penetrations from 4% to 3%.

Evaluation: Testing with Auxiliary Building Ventilation System filtered exhaust in a single train alignment will provide data to ensure the continued acceptable operation of the system. This procedure tests the system in four different configurations. Testing with Auxiliary Building Ventilation System filtered exhaust in a single train alignment will ensure that the Auxiliary Building Ventilation System filter units are tested in their most challenging configuration. The airflow during single train alignment will be higher, on a per train basis, than that achieved during dual train alignment. The filter unit efficiency will be tested in an alignment that provides the most severe challenge to components within the filter units such as the HEPA filters and carbon adsorber. Other ventilation filtered exhaust systems (Control Room Ventilation, Fuel Pool Ventilation, and Annulus Ventilation) also test the filters in a single train alignment. The Auxiliary Building Ventilation System single train alignment for testing being done under this procedure will continue to meet the testing guidelines provided by ANSI N510-80. The same test methods and instruments that are currently used to test the system under dual train operation will be used in the single train configurations with the exception that B-Train flow measurements will be taken at a pitot traverse downstream of the current flow monitor. This is similar to the way that A-Train flow measurements are currently taken. Additionally, three pitot traverses will be taken during several of the configurations to provide evidence of the consistency of the pitot testing method.

It is acceptable for the air flow through a single Auxiliary Building Ventilation System filter train operating in a single train alignment to be above the Technical Specification limit (33,000 cfm) since that limit only applies when the system is in the alignment specified in PT/0/A/4450/001C. However, if the flow rate exceeds the carbon bed residence time limit of 35,340 cfm, the train must be declared inoperable. The procedure has adequate guidance to ensure that, should the flow exceed that limit, the affected train will be declared inoperable.

The procedure requires the opposite train of Auxiliary Building Ventilation, as well as the opposite unit of Auxiliary Building Ventilation and all Fuel Pool Ventilation to be shutdown during part of the testing. Having portions of Auxiliary Building Ventilation System off is acceptable from a nuclear safety perspective since any safety injection signal will override the manual controls and start the filtered exhaust fans in their accident alignment. This ensures that the basic design function of the Auxiliary Building Ventilation Systems, which is to keep the ECCS pump rooms at a negative pressure, will not be affected by these procedures. Also, any alignments imposed by these procedures will not impact the accident flowrates of the Auxiliary Building Ventilation System. The accident flowrates are well below the normal flowrates that are affected by these procedures. The accident flowrates are approximately 6,500 cfm and the system will continue to be able to provide that flowrate throughout the testing. The normal ventilation function of the Auxiliary Building Ventilation System will be temporarily reduced by use of these procedures when Auxiliary Building Ventilation trains are shut down. However, since this is only a temporary condition and the procedure has guidance to return the system to its original configuration if necessary, there will be no Environmental Qualification impact. There is no unreviewed safety question associated with this procedure. No Technical Specification changes are required. No UFSAR changes are required.

181 **Type:** Procedure **Unit:** 1

Title: Procedure TT/1/B/9100/71, Fourth Condenser Circulating Water System Pump Performance Test

Description: This procedure will gather Unit 1 plant data to assess the thermal performance benefits of running a fourth Condenser Circulating Water System Pump. Initially, thermal performance data will be gathered for thirty minutes. Then a fourth Condenser Circulating Water System Pump will be started. Thermal performance data will then be gathered for at least thirty minutes to assess the benefits and Condenser Circulating Water System response to running all four pumps.

Evaluation: Running a fourth Condenser Circulating Water System pump will increase Condenser Circulating Water System flow from 630,000 gallons per minute (gpm) to 760,000 gpm. This is based on curves supplied by the pump manufacturer and was confirmed by an independent analysis performed by Catawba Engineering. This analysis has also shown that the Condenser Circulating Water System Pumps will have adequate Net Positive Suction Head (NPSH) during this mode. Also, a system flow of 760,000 gpm ensures that the minimum flow of 175,000 gpm per pump is maintained with margin. Pump discharge pressure will remain well below the piping design pressure. Condenser Circulating Water System pipe breaks experienced in the past were due not to overpressure, but were due to high residual stresses in the piping during cold weather after a reactor trip.

The cooling towers are designed to take the flow of four Condenser Circulating Water System pumps. The increased flow will raise level in the upper basin approximately three inches. When the fourth pump is started, a drop in the lower basin level is expected as the water inventory shifts from the bottom of the tower to the top. Previous experience has shown that these level disturbances during pump swaps are minimal.

With the higher flow rate, the temperature rise across the condenser will decrease from approximately 24.5 degrees F. to 19.5 degrees F. Based on cooling tower performance curves, cooling tower outlet temperature should increase 1.5 degrees F above the temperature with three pumps running.

Condenser waterbox differential pressure will increase with the higher flow to 5.4 psid per waterbox.

The fourth Condenser Circulating Water System Pump will increase auxiliary electrical load by 4.7 megawatts (MW). Depending on the Condenser Circulating Water System temperature at the beginning of the test, net generation may increase as much as 8 MW. Condensate temperature should decrease by approximately 5 degrees F and vacuum on all three condensers will increase.

The test will be terminated immediately and the fourth Condenser Circulating Water System Pump will be secured if any of the following situations occur:

1 . A steady stream of water from the towers reaching the ground. Any spillover from the top of the towers that is caught by the louvers and directed back into the towers is acceptable. This has been confirmed environmentally acceptable by Catawba Environmental Management.

2. Condenser Circulating Water System pump suction pressure in the red range as indicated on the control board.
3. Condenser Circulating Water System pump flow drops below the minimum pump flow of 175,000 gpm.

Thermal performance models of running four Condenser Circulating Water System pumps indicate that the approximate 5 degree F drop in condensate temperature will have no effect on reactor thermal power.

This activity will have no effect on any of the accidents analyzed in the UFSAR. There will be no effect on the probability or consequences of the analyzed accidents and no new accident scenarios will be created. There are no unreviewed safety questions associated with this procedure. No Technical Specifications changes are required. No UFSAR changes are required.

174 **Type:** Procedure **Unit:** 1

Title: Procedure TT/1/B/9100/71, Fourth Condenser Circulating Water System Pump Performance Test

Description: This procedure will gather Unit 1 plant data to assess the thermal performance benefits of running a fourth Condenser Circulating Water System Pump. Initially, thermal performance data will be gathered for thirty minutes. Then a fourth Condenser Circulating Water System Pump will be started. Thermal performance data will then be gathered for at least thirty minutes. This procedure does not perform any equipment manipulations. All equipment manipulations will be performed by existing plant procedures.

Evaluation: Four Condenser Circulating Water System Pumps are occasionally run during equipment rotation. This procedure extends the time that four pumps are allowed to operate. This activity has no effect on any accident evaluated in the UFSAR. The probability or consequences of accidents evaluated in the UFSAR will not be affected. No new accident scenarios will be created. There are no unreviewed safety questions associated with this procedure. No Technical Specification changes are required. No UFSAR changes are required.

175 **Type:** Procedure **Unit:** 2

Title: Procedure TT/2/B/9100/71, Fourth Condenser Circulating Water System Pump Performance Test

Description: This procedure will gather Unit 2 plant data to assess the thermal performance benefits of running a fourth Condenser Circulating Water System Pump. Initially, thermal performance data will be gathered for thirty minutes. Then a fourth Condenser Circulating Water System Pump will be started. Thermal performance data will then be gathered for at least thirty minutes. This procedure does not perform any equipment manipulations. All equipment manipulations will be performed by existing plant procedures.

Evaluation: Four Condenser Circulating Water System Pumps are occasionally run during equipment rotation. This procedure extends the time that four pumps are allowed to operate. This activity has no effect on any accident evaluated in the UFSAR. The probability or consequences of accidents evaluated in the UFSAR will not be affected. No new accident scenarios will be created. There are no unreviewed safety questions associated with this procedure. No Technical Specification changes are required. No UFSAR changes are required.

70 **Type:** Procedure **Unit:** 0

Title: Procedures PT/0/A/4400/022A and PT/0/A/4400/022B

Description: Procedures PT/0/A/4400/022A and PT/0/A/4400/022B which are the Nuclear Service Water Pump performance periodic tests are being changed to revise the method of calculating the "Total Head". These procedures currently calculate the Total Head as a differential pressure which is incorrect. The differential pressure for most pumps is typically calculated by subtracting the suction pressure from the discharge pressure. The total head for vertical suction pumps drawing water from an open sump (as the Catawba Nuclear Service Water Pumps do) is calculated by measuring the distance from the pump discharge centerline to the free water level in the sump, converting to pressure, then adding discharge centerline to to the free water level in the sump, converting to pressure, then adding to the discharge pressure. This accounts for the work done by the pump to lift the water to the pump discharge. This change will revise procedure enclosures to calculate the total head correctly and rename the calculated value as "Total Head".

Evaluation: This change does not affect the facility as described in the UFSAR. Correcting this procedure to correctly make the calculation will not have an effect on the probability or consequence of accidents evaluated in the UFSAR. There are no unreviewed safety questions associated with this procedure change. No Technical Specification changes are required. No UFSAR changes are required.

124 **Type:** Procedure **Unit:** 1

Title: PT/1/A/4400/014, Revision 3, Nuclear Service Water System to Auxiliary Feedwater System Suction Piping Flow Measurement

Description: Procedure PT/1/A/4400/014, Nuclear Service Water System to Auxiliary Feedwater System Suction Piping Flow Measurement, establishes flow through the Nuclear Service Water System to Auxiliary Feedwater System Suction piping to measure flow and pressure drop in the piping in order to determine if the piping roughness is acceptable for the Auxiliary Feedwater System to meet its design basis requirements. The procedure presently requires the test equipment to be installed and then later to have the same test equipment vented. Changes being made to this procedure in revision 3 allow the installation and venting of test equipment to occur sequentially in the same step. A step is also being added to direct Operations personnel to perform the Nuclear Service Water System to Auxiliary Feedwater System suction piping flush per procedure PT/1/A/4200/059 if desired. This step allows for the completion of the monthly required flush following testing and is added to reduce the administrative tasks associated with removing Auxiliary Feedwater System from service and returning it to service for two similar procedures. The piping flush procedure is an independent procedure which has been previously evaluated. Successful completion of this test is not contingent upon the performance of the flush procedure and the step to have this procedure perform is added only for convenience. Additional steps are being added to ensure the strainer basket in the Nuclear Service Water System flush line is removed prior to the test and reinstalled following the test to ensure a discharge flow path exists during the performance of the test.

Evaluation: There are no unreviewed safety questions involved with this change to PT/1/A/4400/014, Nuclear Service Water System to Auxiliary Feedwater System Suction Piping Flow Measurement. Neither the Nuclear Service Water System nor the Auxiliary Feedwater System are accident initiators of the accidents analyzed in the UFSAR. These procedure changes 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.

223 **Type:** UFSAR Change **Unit:** 0

Title: Change to UFSAR Section 1.8.1.30 concerning the Operator Aid Computer

Description: A discrepancy was identified in UFSAR Section 1.8.1.30 about the Operator Aid Computer Incore Thermocouple mapping capability. The UFSAR was not updated following the Operator Aid Computer replacement which was performed per Nuclear Station Modifications CN-11329 for Unit 1 and CN-21329 for Unit 2, to reflect the fact that the new Operator Aid Computer thermocouple map only displays temperature and does not display "enthalpy rise". UFSAR Section 1.8.1.30 describes various Inadequate Core Cooling Instruments mandated by NUREG 0737. The incore thermocouple system is one of these instruments. It provides the operator with core exit temperatures at 65 core locations. It provides inputs to the Operator Aid Computer and to the Inadeqaute Core Cooling Monitor for subcooling margin calculations and incore thermocouple mapping applications. The former Honeywell Operator Aid Computer had a provision to calculate the approximate enthalpy rise at each thermocouple location. This feature is not available with the new Operator Aid Computer.

Evaluation: There are no unreviewed safety questions associated with this UFSAR change. The Operator Aid Computer display of incore thermocouple system data is not safety related. No credit is taken for it in accident analyses. No Technical Specification changes are required. UFSAR Section 1.8.1.30 will be revised.

63 **Type:** UFSAR Change **Unit:** 0

Title: Change to UFSAR Section 9.2.2.6.2

Description: UFSAR Section 9.2.2.6.2 currently indicates that the Component Cooling System drain sump level is given locally for sump level indication. Actually there is no local indication.

Evaluation: There is no unreviewed safety question associated with this UFSAR change. This change revises the UFSAR to agree with system design drawings. Two level switches monitor the Component Cooling Drain Sump water level. These switches start and stop the two sump pumps and provide high-high and low-low level alarms. No Technical Specification changes are required. No UFSAR changes are required.

31 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Chapter 16, Selected Licensee Commitments Section 16.11.16.2 Annual Radioactive Effluents Release Report

Description: In Selected Licensee Commitment 16.11-16.2 which addresses the Radioactive Effluent Release Report, the method of accounting for meteorological conditions is being changed. The following sentence will be deleted: "The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses." The following sentences will be inserted. "A five-year average of representative onsite meteorological data shall be used in rhe gaseous effluent dose pathway calculations. Dispersion factors and desposition factors shall be generated using the computer code XOQDOQ (NUREG/CR-2919) which implements NRC Regulatory Guide 1.111. The meteorological conditions concurrent with the time of release shall be reviewed annually to determine if the five year average values should be revised."

Evaluation: There are no unreviewed safety questions associated with this UFSAR Change. Calculations using meteorological data as a part of the determination of radioactive materials in gaseous effluents do not affect the probability of an accident. NUREG-0133 clearly allows use of historical annual average meteorological data for calculating gaseous effluent dose pathway calculations for both "long term" and "short term" releases. No Technical Specification changes are required. A change is required for Selected Licensee Commitments Section 16.11.16.2.

145 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Chapter 7, UFSAR Table 7-12 and UFSAR Figure 7-2 related to removal of Remote Load Dispatching

Description: The UFSAR in Table 7-12 and Figure 7-2 (Part 9 and 16) refer to an option, "Remote Load Dispatching", that does not exist at Catawba. This was an option offered by Westinghouse and shown on their standard drawings as an option, but this option was never accepted or implemented at Catawba. The UFSAR references to this option will be removed, but no plant equipment or plant operation is affected by this change

Evaluation: This UFSAR change removes inaccurate technical information. This change has no effect on the operation, design bases, or function of any structure, system or component. The corrections or changes do not involve any changes to the operation, design basis or function of any structure, system or component (SSC). No safety or licensing issues are involved and no revisions to regulatory commitments are involved by the corrections or changes. There is no unreviewed safety question associated with this change. No Technical Specification changes are required. A change is required for UFSAR Chapter 7, UFSAR Table 7-12 and UFSAR Figure 7-2.

136 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 1.8.1.34.15

Description: UFSAR Section 1.8.1.34.15 describes the methods used to identify gross leakage from systems located outside of containment that contain primary coolant. This section currently lists "Radiation Protection surveys and proper system operation" as methods that are used to verify that gross leakage is not occurring from systems containing primary coolant outside containment. This UFSAR change will add "tank and sump monitoring" and "operator rounds" to more accurately describe the methods by which Catawba verifies that gross leakage is not occurring from these systems.

Evaluation: There is no unreviewed safety question associated with this UFSAR change. Since no physical or operational changes will be made to any system, structure, or component; this is basically an editorial change. There will be no effect on the probability or consequences of accident evaluated in the UFSAR. No Technical Specification changes are required. UFSAR Section 1.8.1.34.15 will be revised.

226 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 10.4.9.2 to reflect Motor Driven Auxiliary Feedwater Pump start on AMSAC

Description: UFSAR Section 10.4.9.2 was revised to reflect Motor Driven Auxiliary Feedwater Pump start on AMSAC. The ATWS Mitigation System Actuation Circuitry (AMSAC) System was installed previously per Station Modifications CN-10952 (Unit 1) and 20952 (Unit 2). The UFSAR revision was not made at the time the modifications were implemented. This UFSAR change is being made to show the as-built condition of the plant.

Evaluation: There are no actual plant changes associated with this UFSAR change. The change is being made to incorporate changes that should have been made when the AMSAC modification was installed. There are no unreviewed safety questions associated with this UFSAR change. No Technical Specification changes are required. UFSAR Section 10.4.9.2 will be revised.

36 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 11.2

Description: UFSAR Section 11.2 is being changed to correct numerous inaccuracies regarding the operation of the Liquid Radwaste Recycle System. In the years since initial plant start-up, the processing and release of radioactive waste has been optimized and improved. The majority of these inaccuracies involve the fact that the Liquid Radwaste Recycle System Evaporator has never been used and Monitor Tank Building processing equipment has been added. Operation of the system will continue such that the amounts of radioactive material in liquid effluents are reduced to assure that doses to individuals beyond the site boundary are within the limits of 10CFR50 Appendix I. Applicable Selected Licensee Commitments and Regulatory Guide 1.21 will continue to be adhered to. This UFSAR change resulted in changes to Operations Procedures OP/1/A/6500/014 (Revision 55) and OP/2/A/6500/14 (Revision 43).

Evaluation: In general, none of the equipment used to process and release Liquid Radwaste Recycle System water are UFSAR Chapter 15 accident initiators. The process changes described in this package do not affect the intended function, operation, or reliability of the safety related Residual Heat Removal System, Containment Spray System, and Auxiliary Feedwater System sump pumps. There are no unreviewed safety questions associated with this UFSAR change. No Technical Specification changes are required. Extensive changes are required for UFSAR Section 11.2.

35 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 11.2.2.7.2.2

Description: This change to UFSAR Section 11.2.2.7.2.2 corrects inaccuracies concerning the role of the incore instrumentation room sump with respect to reactor building leakage detection and the methods by which the containment floor and equipment sumps are used to identify and quantify leakage per the requirements of Regulatory Guide 1.45. The section states that "the containment floor and equipment sump pumps as well as the incore instrumentation room sump pump, input to a plant computer program designed to detect one gallon per minute of unidentified leakage inside containment in less than one hour as required by NRC Regulatory Guide 1.45." This statement is incorrect because the incore instrumentation room sump does not provide input to the Operator Aid Computer Program which calculates containment floor and equipment sump leakage input rate. Section 11.2.2.7.2.2 also inaccurately states that containment floor and equipment sump pump run time is used to determine the sump leakrate and that an alarm is provided if the leakrate is greater than one gallon per minute. There are no containment sump leakage alarms that are generated based on sump pump run time. Actually, the Operator Aid Computer is programmed to provide gross leakage alarms at input rates of 3, 15, and 50 gallons per minute. This UFSAR change is essentially editorial since no physical or operational changes will be made to any system, structure or component. This UFSAR change will not affect the ability of the sump to meet the requirements of Technical Specification 3.4.15 or Regulatory Guide 1.45.

Evaluation: There are no unreviewed safety questions associated with this UFSAR Change. No Technical Specification changes are required. A revision is required for UFSAR Section 11.2.2.7.2.2.

169 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 15.6.5.3

Description: UFSAR Section 15.6.5.3 was revised to report that credit is taken for the Auxiliary Building Ventilation System to limit the release of radioactivity to the environment following a design basis LOCA with leakage from the Engineered Safety Features (ESF) Pumps. Values assumed for the efficiencies of the filters of the Auxiliary Building Ventilation Systems for the removal of iodine are reported. In addition, the current analyses of radiological consequences of design basis events are revised to resolve a number of concerns recently identified. Notes were added in section 15.6.3 and associated tables.

Evaluation: The statements concerning credit for the Auxiliary Building Ventilation System and the values reported for the assumed efficiencies of its filters reflect the assumptions made in the current analyses of radiological consequences of a LOCA. There are no unreviewed safety questions associated with this revision. No Technical Specification changes are required. A change is required for UFSAR Section 15.6.5.3.

28 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 16.7-10, Radiation Monitoring for Plant Operations

Description: UFSAR Section 16.7-10, Radiation Monitoring for Plant Operations, was changed to revise the wording of Action "H" Table 16.7-10A. In July 1998 Technical Specification 3/4.3.3 was being reviewed in preparation for transition to the Improved Technical Specifications. Technical Specification 3/4.3.3 addressed radiation monitoring for plant operations. It contained lower limit of detection (gross gamma) requirements for the Component Cooling Water System. The methodology used to measure lower limit of detection values involves measuring values for specific radioisotopes rather than measuring a gross gamma value. Technical Specification 3/4.3.3 became Selected Licensee Commitment 16.7-10 after transistion to the Improved Technical Specifications. To correct this problem, Selected Licensee Commitment Table 16.7-10, Remedial Action Statement H was revised to state "With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, operation may continue for up to 30 days provided that at least once per 12 hours, grab samples are collected and analyzed for Principle Gamma Emitters (listed in Table 16.11-1, Table Notation 3) at a lower limit of detection no more than 5x10E-7 microcuries per milliliter.

Evaluation: Current Selected Licensee Commitment requirements for Liquid Effluent and Process samples and associated lower limits of detection are derived from NRC Regulatory Guide 1.21 (Revision 1, June 1974) - "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light Water-Cooled Nuclear Power Plants", Appendix A, Section B. This section describes the sensitivity to which liquid effluents should be analyzed prior to release from site. The Component Cooling System Grab Samples which are collected are referenced in UFSAR 11.5.1.2.1.5 as Process and Effluent Samples. The Component Cooling System is a closed system and therefore the Radiation Monitors are monitoring a liquid process type system, not an effluent type system. Regulatory Guide 1.21 states :" When operational or other limits preclude specific gamma radionuclide analysis of each batch, gross radioactivity measurements should be made to estimate the quantity and concentrations of radioactive material released in the batch....". The Component Cooling System grab samples are not effluent and are not batch released samples, therefore the next level of analysis would be for "specific gamma radionuclides". The Regulatory Guide further states "...sensitivities of analyses of radioactive materials in liquid effluents should be sufficient to permit the measurement of concentrations of 10E-7 microcuries/milliliter by gross radioactivity measurements, 5x10E-7 microcuries/milliliter of each gamma-emitting radionuclide,..." therefore, this change will require that the liquid samples taken from the Component Cooling System will be analyzed as other liquid waste release/ effluent samples to the same lower limit of detection. The gross radioactivity measurement is not necessary because current methodology allows the ability to accurately detect liquid effluent radioactivity for specific gamma emitters to the 5x10E-7 microcuries/milliliter lower limit of detection.
This change will not have any effect on accidents analyzed in the UFSAR or create an accident that has not been analyzed in the UFSAR. There are no unreviewed safety questions associated with this change. No Technical Specification changes are required. A revision is required for UFSAR Section 16.7-10.

90 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 2.2.3.1.4

Description: UFSAR Section 2.2.3.1.4 was revised to describe the location of toxic gases at the plant. This section of the UFSAR describes the potential impact of toxic gases on the plant. The original design of Catawba Nuclear Station utilized hypochlorite and sulfuric acid for water treatment and other purposes. These chemicals were both stored in tanks inside the Water Chemistry Building. No gaseous chlorine was needed on site. This original design is currently described in the UFSAR. The station has been modified to have gaseous chlorine at several locations around the site and the hypochlorite storage have been relocated. The location of sulfuric acid storage has been chnaged as well.

Evaluation: The changes to the UFSAR were all made to reflect the current condition of the station. The description of the new location for the storage of hypochlorite does not raise any safety concerns since it has been moved away from any source of sulfuric acid. The original concern was that hypochlorite and sulfuric acid could combine to form chlorine gas. The potential for an accident of this nature is lower than the original design since the bulk storage of these two chemicals are no longer in close proximity. Minor modification CE-60240 controlled the relocation of the hypochlorite tanks. Sodium hydroxide storage is currently being changed. This will be handled through the normal modification process. No new storage locations for this chemical are being made therefore no additional risks are created. The locations of onsite gaseous chlorine have been evaluated and documented in calculation CNC-1211.00-00-0047. Since the station is still in compliance with Regulatory Guides 1.95 and 1.78, no new safety concerns have been created by the use of gaseous chlorine. Other changes to UFSAR section 2.2.3.1.4 are considered editorial and do not require further evaluation. None of these changes affect the probability or consequences of accidents evaluated in the UFSAR. No new accident scenarios are created. The Control Room is protected from chlorine by chlorine detectors which close the intakes to the Control Room Ventilation System upon detection of elevated chlorine concentration at the air intakes. There are no unreviewed safety questions associated with this UFSAR Change. No Technical Specification changes are required. UFSAR Section 2.2.3.1.4 will be revised.

58 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 2.3.1.2 and Table 2-27

Description: The activity covered by this evaluation is a revision to UFSAR Section 2.3.1.2 (Regional Meteorological Conditions for Design and Operating Bases) and Table 2-27 (Tropical Cyclones Affecting the Site Area). For the first item slight changes in the maximum rainfall and maximum wind speeds were necessary because of meteorological events that have occurred over the last several years. For the second item three tropical storms (occurring over the last several years) were added to the list of tropical storms in the UFSAR Table. In addition a few minor editorial changes were made associated with the above updates (mainly to indicate that the information came from the National Weather Service). A small clarification is also made on the time period for tornado data that is given in the UFSAR

Evaluation: There are no unreviewed safety questions associated with this UFSAR change. The changes to the UFSAR include addition to Section 2.3.1.2 of recent historical information on the inland maximum wind speed and rainfall and an addition to Table 2-27 to document tropical cyclones (hurricanes and a tropical storm) that occurred within the last several years. The previous maximum wind speed documented for inland areas of the region where Catawba Nuclear Station is located was 80 miles per hour occurring at Wilmington, NC. The current maximum wind speed recorded is 87 miles per hour at Charleston, SC (September 22, 1989). The Nuclear Safety Related Structures at Carawba Nuclear Station are designed for a maximum wind speed of 95 miles per hour (for conditions other than a tornado for which the design wind speed is higher). The existing design and analysis envelopes the actual wind speed that has been experienced regionally. The new maximum inland rainfall in a 24 hour period is 9.2 inches at Greenville-Spartanburg, SC on August 25, 1995. This value for maximum rainfall is well below the maximum rainfall for which the station yard drainage is analyzed. The changes for this proposed UFSAR revision merely provide updated information. There is no change to the design or design basis of any plant structure, system, or component related to natural phenomena required in conjunction with the proposed UFSAR revision. There are no operating characteristics, failure modes and effects, or accident analyses that are affected by the information provided in this proposed UFSAR revision. The probability of an accident occurring at Catawba is not affected at all by the fact that slightly higher rainfall and wind occurred in the geographical region where Catawba is located than had previously been documented. In addition, natural phenomena are not postulated to cause accidents described in the SAR. Equipment important to safety is protected from natural phenomena (such as those evaluated here) by QA Condition I structures. Such structures are designed and analyzed for conditions that envelope those that are described and quantified in this UFSAR revision. No Technical Specification changes are required. Changes are required for UFSAR Section 2.3.1.2 and Table 2-27.

34 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 4.2.1.2.1

Description: Calculation DPC-1553.26-0093, Revision 2 contains a generic 10CFR50.59 analysis and safety review for the coarse and fine mesh bottom filter nozzles at Catawba. This analysis evaluated whether there were any Unreviewed Safety Questions (USQ) associated with the bottom nozzle designs. The following discussion summarizes the results of the evaluation:

Normal Operation

During normal operation, the particulate size transported from the Reactor Coolant System filters to the core is smaller than the bottom nozzle holes per Framatome Cogema Fuels (FCF) calculation 32-5003654-00, Mk-BW Fine Mesh Fuel Filter LOCA Impact Evaluation. Therefore, blockage of the bottom nozzles is not expected. Therefore, no new failure modes have been introduced. Therefore, the probability of an accident that is different than previously evaluated in the SAR is not increased.

Post LOCA Conditions Blockage of fuel assembly channels is not credible for a cold leg break LOCA per FCF calculation 32-5003654-00 and Westinghouse document, "Effect Of Debris During Post Accident Conditions On Cores With Debris Resistant Bottom Nozzles". FCF calculation 32-5003654-00 states that flow velocities are insufficient to entrain or lift debris into the core inlet. Thus, there will be no significant debris transport to the core inlet following a cold leg break LOCA. Only small amounts of any semi-neutral density debris, if present, could be carried to the assembly entrances following a cold break LOCA. However, there are insufficient quantities of buoyant material in containment that could be transported to the core inlet following a cold leg break LOCA. Therefore, core blockage following a cold leg break LOCA is not credible.

In addition, the Westinghouse document independently verified that there are insufficient fluid velocities to transport debris into the core during cold leg break LOCA conditions. Thus, the normal flow path of coolant up through the fuel assemblies following a LOCA is undisturbed. Therefore, the operability and the capability of the ECCS to perform its safety related function (i.e., provide adequate core cooling following a LOCA) are not adversely impacted by this modification.

FCF calculation 32-5003654-00 verified that adequate core cooling can be provided by flow redistribution following a cold break LOCA even if there were limited blockage by debris such as paint chips. Therefore, the capability of the ECCS to provide adequate core cooling following a cold leg break LOCA is not adversely impacted by the modification to the bottom nozzle.

FCF calculation 32-5003654-00 and the Westinghouse document state that for the hot leg break LOCA, the fluid velocities in the lower plenum are only marginally adequate to transport and sustain debris at the core inlet. FCF calculation 32-5003654-00 indicates that debris greater than 7.2 mils thick will not be transported to the core inlet. Based on coating specifications, debris sources smaller than 11 mils (laminated) and 5-8 mils (delaminated) are not credible. Therefore, it is unlikely that any significant blockage of the bottom nozzle would occur during hot leg break conditions and significant blockage would be self-limiting. The FCF calculation 32-5003654-00 shows that, in the unlikely event that the core inlet were completely blocked, more than adequate flow is always available via alternate flow paths through the core basket.

FCF calculation 32-5003654-00 demonstrates that the operability of the Emergency Core Cooling system is not adversely affected by the modification to the bottom nozzle. This calculation verifies that adequate core cooling can be provided via alternate flow paths following a hot leg break LOCA even if the core were completely blocked by debris from the containment sump. Therefore, the capability of the ECCS to perform its safety related functions are not adversely affected by this modification.

In addition, FCF calculation 32-5003654-00 states that the grid/fuel pin tolerances, as small as zero, are limiting. Normal spacer grids have zero gap at all grid bumper locations. These tolerances existed in previous and current fuel designs at Catawba before the introduction of the fine mesh bottom filter nozzle. Therefore, the fine mesh bottom filter nozzle does not increase the likelihood of fuel assembly blockage.

Since there is no potential for loss of cooling capability during hot and cold leg break LOCA conditions and the likelihood of fuel assembly blockage is not increased, there are no new failure modes introduced. FCF calculation 32-5003654-00 states that the fuel assembly fine mesh bottom filter size is not limiting in regards to particulate passage tolerance. Grid/fuel pin tolerances, as small as zero, for McGuire and Catawba units, were limiting before the introduction of the fine mesh fuel filter. Therefore, the grid/fuel pin tolerances remain limiting. Thus, FCF Calculation 32-5003654-00 concluded that the Mark-BW fuel assemblies are no more or no less subject to trapping debris as a result of incorporating the debris filter bottom nozzle design described in revision 0 of this calculation. Therefore, there is no LOCA impact due to this modification.

There is no increase in the probability of a malfunction of equipment important to safety previously evaluated in the SAR. Similarly, there is no increase in the probability of a malfunction different than previously evaluated in the SAR. Thus, it is concluded that this modification does not adversely affect the capability of the ECCS to mitigate the consequences of a design basis accident. Therefore, there is no increase in the consequences from an accident previously evaluated in the SAR.

FCF calculation 32-5003654-00 verifies that the operability and the capability of the ECCS system to perform its safety function (i.e., provide adequate core cooling following a LOCA) is not adversely affected by this modification. Therefore, the capabilities of the ECCS system to mitigate the consequences of a LOCA are not adversely impacted. Adequate core cooling is provided by the ECCS system following a LOCA. By ensuring adequate core cooling via the ECCS system, (1) the number of failed fuel rods is minimized and (2) all 50.46 criteria are met. Therefore, this modification does not increase the dose consequences from an accident previously evaluated in the SAR.

Margin of Safety

Improved Technical Specification (ITS) Section 3.5.2 and B 3.5.2, Emergency Core Cooling System (ECCS) and Bases: These specifications state that two trains of ECCS must be operable. As discussed above, this modification does not adversely affect the operability of ECCS or its capability to perform its safety related functions. Therefore, the margin of safety is not reduced.

ITS Section 3.4.16 and B 3.4.16, RCS Specific Activity and Bases: This specification states the acceptable limits for the specific activity in the Reactor Coolant System during normal operation. As discussed above, this modification does not adversely affect the mechanical or thermal performance of the fuel or its reliability during normal operation. Therefore, the margin of safety is not reduced.

ITS Section 3.4.1 and B 3.4.1, Pressure, Temperature and Flow Departure from Nucleate Boiling Limits and Bases: This specification specifies the DNB parameters that ensure that the fuel does not undergo departure from nucleate boiling. Adherence to these parameters ensures the appropriate thermal performance of the fuel during operation. As discussed above, this modification does not adversely affect the mechanical or thermal performance of the fuel and or its reliability. Therefore, the margin of safety is not reduced.

Evaluation: There are no Unreviewed Safety Questions or changes to the Improved Technical Specifications associated with this modification. Changes are required to UFSAR Section 4.2.1.2.1.

201 **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 currently in the UFSAR was previously changed to insert the correct pump curve. The current 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 the result of flow separation at the pumps diffuser and does not introduce any operational restrictions, since the pump operates at 100 percent flow.

Evaluation: There are no unreviewed safety questions associated with this UFSAR change. The change has no actual effect on plant systems, structures or components. No Technical Specification changes are required. UFSAR Section 5.4.1.3.1 will be revised.

206 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 5.4.1.3.1

Description: UFSAR Section 5.4.1.3.1, third paragraph (addressing reactor coolant pump performance) 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." The words "low power" will be added between "during" and "operation" since this situation is only applicable below 48% reactor power operation. The loss of a reactor coolant pump will trip the reactor at power levels greater than 48%.

Evaluation: There are no unreviewed safety questions associated with this UFSAR change. The change is a clarification and has no effect on the actual operation of the plant. 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.

245 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 5.4.1.3.1

Description: UFSAR Section 5.4.1.3.1 which addresses reactor coolant pumps states that "The Reactor Trip System ensures that pump operation is within the assumptions used for loss of coolant flow analyses, 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 sentence will be replaced with the following sentence: "The Reactor Trip System ensures that pump operation is within the assumptions used for loss of coolant flow analyses."

Evaluation: There are no unreviewed safety questions associated with this UFSAR change. This change has no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. UFSAR Section 5.4.1.3.1 will be revised.

49 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 5.4.10.4

Description: UFSAR Section 5.4.10.4 is being changed to remove "manway attachment welds" from the list of welds that are designed and constructed to present a smooth transition surface between the parent metal and the weld metal and which are ground smooth for ultrasonic inspection.

Evaluation: This changes involves deleting inaccurate technical information. There is no change to the operation of the plant or to the design basis or to the function of the affected systems, structures and components. There is no unreviewed safety question associated with this change. No Technical Specification changes are required. A change is required for UFSAR Section 5.4.10.4.

50 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 5.4.10.4

Description: UFSAR Section 5.4.10.4 is being revised to change "all girth and longitudnal full penetration welds" to "Girth and longitudnal full penetration welds". This is in the list of welds that are designed and constructed to present a smooth transition surface between the parent metal and the weld metal and which are ground smooth for ultrasonic inspection.

Evaluation: This changes involves deleting inaccurate technical information. There is no change to the operation of the plant or to the design basis or to the function of the affected systems, structures and components. There is no unreviewed safety question associated with this change. No Technical Specification changes are required. A change is required for UFSAR Section 5.4.10.4.

198 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 5.4.11.5

Description: UFSAR Section 5.4.11.5 states that the "Pressurizer Relief Discharge System" components are tested in accordance with Section VIII of the ASME Code". This is accurate for the Pressurizer Relief Tank but the remainder of the system is tested in accordance with ANSI B 31.1 Power Piping Code. This UFSAR reference is for construction testing. UFSAR Section 5.4.11.5 will be revised to state the correct code.

Evaluation: There is no unreviewed safety question associated with this UFSAR change. The change corrects erroneous information. This change has no effect on plant systems, structures, or components. The change has no effect on any accidents analyzed in the UFSAR and no new accident scenarios are created. No Technical Specification changes are required. UFSAR Section 5.4.11.5 will be revised.

51 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 5.4.11.5

Description: UFSAR Section 5.4.11.5 was revised to replace the term "system components" with "Pressurizer Relief Tank". A sentence was added to indicate that system piping and valves are constructed and tested in accordance with the requirements of ANSI B31.1"

Evaluation: This changes involves adding technical information for clarification. There is no change to the operation of the plant or to the design basis or to the function of the affected systems, structures and components. There is no unreviewed safety question associated with this change. No Technical Specification changes are required. A change is required for UFSAR Section 5.4.11.5.

52 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 5.4.12.4

Description: UFSAR Section 5.4.12.4 was revised to change the applicable code from ASME Section III to ASME Section XI.

Evaluation: This change involves revising an incorrect piece of technical information. There is no change to the operation of the plant or to the design basis or to the function of the affected systems, structures and components. There is no unreviewed safety question associated with this change. No Technical Specification changes are required. A change is required for UFSAR Section 5.4.12.4.

53 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 5.4.13.4

Description: This change involves noting that the Main Steam Safety Valves are tested with steam instead of air.

Evaluation: This changes involves deleting inaccurate technical information. There is no change to the operation of the plant or to the design basis or to the function of the affected systems, structures and components. There is no unreviewed safety question associated with this change. No Technical Specification changes are required. A change is required for UFSAR Section 5.4.13.4.

199 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 5.4.13.4

Description: UFSAR Section 5.4.13.4 which addresses Tests and Inspections states that the Pressurizer Safety Valve setpoints are set with air. The setpoints are actually set with steam.

Evaluation: There is no unreviewed safety question associated with this UFSAR change. This change has no effect on plant systems, structures, or components. The change has no effect on any accidents analyzed in the UFSAR and no new accident scenarios are created. No Technical Specification changes are required. UFSAR Section 5.4.13.4 will be revised.

30 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 5.4.7, Residual Heat Removal System

Description: UFSAR Section 5.4.7.1, Residual Heat Removal System "Design Bases", UFSAR Sections 5.4.7.2.3 and 5.4.7.3 are being changed and UFSAR Figures 5-19 and 5-20 are being deleted. These changes are editorial and serve to clarify existing design capability of the Residual Heat Removal System to achieve cold shutdown. The change also eliminates conflicts between this section and actual plant experience. Current analyses demonstrate that with only one Residual Heat Removal train in service, no reactor coolant pumps operating, and with the heat exchanger supplied with component cooling water at design flow and temperature, the Residual Heat Removal System is capable of reducing the temperature of the reactor coolant from 350 °F to 200 °F with decay heat at 20 hours after reactor shutdown. Thus, single failure design basis capability has been demonstrated consistent with the original design analysis. There is no cooldown related safety analysis requirement or acceptance criteria associated with maintaining temperature any less than 200 °F. Mode 5 Cold Shutdown (<200 °F) is the required safety state for all plant Technical Specifications requiring shutdown from operating conditions. The performance of components or systems will not be degraded by the change. The margins present in the ITS 3.4.3 NC System Heatup and Cooldown Curves are not affected by this change. These curves (along with Low Temperature Overpressure requirements per ITS 3.4.12) impose the necessary conservative limits based on the limiting materials properties of the reactor pressure boundary.

Evaluation: There are no physical changes to the plant associated with these UFSAR revisions. There are no unreviewed safety questions associated with these UFSAR changes. No changes to any Technical Specification are required. Changes are required for UFSAR Sections 5.4.7.1, 5.4.7.2.3, 5.4.7.3 and UFSAR Figures 5-19 and 5-20.

25 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 6.2.1.1.3, Loss of Coolant Accident

Description: UFSAR Section 6.2.1.1.3 is being changed to revise the known ice condenser bypass leakage from 2.2 square feet to 2.3 square feet. This is based on the existence of two drains in the floor of the Containment Air Return Fan Pit at elevation 593 feet 8.5 inches in both the Unit 1 and Unit 2 Containment. The drains go directly through the floor slab. They have been in existence since the construction of the plant but have never been specifically documented as bypass leakage. The additional documented bypass leakage is well within the allowable leakage of 5.0 square feet and the acceptable leakage based on full scale test results of 40 square feet. No nearby equipment is affected in a detrimental manner under normal operating, accident, or any other conditions. The ice condensers and nearby containment air return fans will still function as designed. There is slightly less chance that both trains of the containment air return fan system would be subject to a common mode failure during a design basis accident. This is a positive benefit of the drains being in place.

Evaluation: There is no unreviewed safety question associated with this UFSAR change. No Technical Specification changes are required. UFSAR Section 6.12.1.1.3 will be revised.

255 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 6.3.2.2 regarding drain path for Cold Leg Accumulators

Description: UFSAR Section 6.3.2.2, fourth paragraph under the heading of Cold Leg Injection Accumulators; incorrectly states that the Cold Leg Accumulators are drained to the Reactor Coolant Drain Tank during level adjustments during normal operation. Operating procedures have drained the Cold Leg Accumulators to the Refueling Water Storage Tank via the Safety Injection System test header since 1982. The Cold Leg Accumulator drain procedure was recently revised to allow Cold Leg Accumulator drain via the Cold Leg Accumulator sample lines. Per the Safety Injection System flow diagrams, the Cold Leg Accumulator drain to the Reactor Coolant Drain Tank were designed to only be used when the Cold Leg Accumulators are depressurized.

Evaluation: The evaluation concludes that the alternate drain configuration for level decrease in the Cold Leg Accumulators involves no unreviewed safety questions. Methods for determining boron concentration are adequate to ensure the boron concentration is maintained within operability limits whether the Cold Leg Accumulator level decrease is established through the Nuclear Sampling System or by the current method through the Safety Injection System test header. No changes to Technical Specifications are required. No UFSAR changes are required by this change,

The Cold Leg Accumulators function along with the other ECCS flows, ensures maximum fuel clad temp, cladding oxidation and hydrogen generation limits are not exceeded and the core is maintained in a coolable geometry following a LOCA. This change does not affect the boron concentration, volume or pressure in the accumulators. Therefore, the assumptions of the LOCA analysis are not affected by this change. This change does not decrease the margin for safety for the cold leg accumulators and no change to the fission product barriers from the assumptions of the UFSAR will result from this change. Draining through the Safety Injection System test header to the Refueling Water Storage Tank does not significantly dilute the Refueling Water Storage Tank even though the allowable boron concentration in the Cold Leg Accumulators is lower than the minimum boron concentration in the Refueling Water Storage Tank. This is based on an assumed maximum Cold Leg Accumulator in-leakage of 1 gpm. No changes to Technical Specifications are required. No UFSAR changes are required by this change.

180 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 6.3.3

Description: Three changes will be made to UFSAR Section 6.3.3. The changes reflect an update in the analysis performed to be consistent with Westinghouse hot leg switchover methodology (NASL-92-010).

Change 1: The mass boil off value will be updated based on 1971 ANS decay heat including 20% uncertainty. This increases the calculated value from 28-30 lbm/sec to 35 lbm/sec.
Change 2: A statement will be added to identify the recommended hot leg injection flow rate. The recommended hot leg injection flow rate is 1.3 times the decay heat boil off rate. Thus the recommended hot leg injection flow rate is 46 lbm/sec.
Change 3: A change will be made to make the Catawba UFSAR consistent with the McGuire UFSAR. The change will indicate that hot leg injection is required at "approximately" 6 hours. This change makes this statement more accurate since, using a conservative method, slightly more than six hours are available to perform this action.

Evaluation: The hot leg injection flow requirements come from 10CFR50.46 (b) acceptance criteria (4) "Coolable geometry" and (5) "Long-term cooling". To maintain a coolable geometry, boron precipitation must be prevented. The recommended hot leg injection flow exceeds the decay heat boil off flow rate and thus provides excess flow to flush the core region. This flushing flow will result in reducing any boron concentration build up that may have been occurring. By providing a flow rate through the hot leg injection path that exceeds the flow rate required to remove decay heat, long term cooling is ensured. The decay heat rate is calculated using the 10CFR Part 50 Appendix K required decay heat and includes actinide decay. It should be noted that the Appendix K required decay heat conservatively bounds the expected decay heat and provides design margin.

Hot leg switchover analysis methods were originally developed in the mid 1970's in response to NRC concerns regarding potential post-LOCA boron precipitation. The analyses were presented to the NRC but not considered as part of the formal "LOCA methods". Minor refinements in the methodology have been made based on a forward-fit bases which preserved the overall approach. Thus the calculated hot leg injection flow rates related to this change use "approved" methods since they are consistent with the approach presented to the NRC.

The are two hot leg injection requirements, hot leg switchover time and hot leg injection flow rate. The switchover time is specified in the Emergency Operating Procedures (EOPs) and the hot leg injection flow rate is specified in the Test Acceptance Criteria (TACs). The hot leg switchover time specified in the EOPs determines the time at which hot leg injection can be implemented. The time is an input to the decay heat calculation. Given the switchover time, a decay heat rate and injection flow requirements can be determined. The required hot leg injection flow rate (system flow requirements) is specified in the TACs.

The hot leg injection flow requirements specified in this revision to the UFSAR, meet the regulatory requirements of 10CFR50.46 and 10CFR50 Appendix K, and are consistent with the Westinghouse methods. In addition, these requirements are consistent with the hot leg switchover time specified in the EOPs and flow requirements specified in the TACs.

There are no unreviewed safety questions associated with this UFSAR change. No Technical Specification changes are required. UFSAR Section 6.3.3 will be revised.

205 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 7.1.2

Description: UFSAR Section 7.1.2, "Identification of Safety Criteria", includes the follwing sentence: "In addition, plant procedures provide assurance that whenever a part of a redundant system is removed from service, the portion remaining in service is functionally tested immediately after disabling of the affected portion". This sentence will be deleted. The deletion will concur with the protection system's "single failure" design intent concerning a channel being out of service. There should be no functional testing allowed on any of the remaining channels when one channel is already out of service during normal operation.

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 7.1.2 will be revised.

89 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 9.1.4.3.1

Description: UFSAR Section 9.1.4.3.1 was changed to revise incorrect information regarding the fuel hoist brake capacities. In this section, it is stated that the fuel hoist motor brake capacity is 350% of operating load and the mechanical load brake at 350 %, also. This information could not be found in any documented source. The manufacturer of the refueling machine was consulted about the validity of this information, and it was found to be incorrect. The manufacturer, Raytheon (formerly Stearns-Roger) designed the fuel handling crane and hoist according to the Crane Manufacturers Association of America (CMAA) Specification 70. This specification requires that the hoist motor brake be designed for 100% rated load when combined with a mechanical load brake, also designed for 100 % of rated load. This change will reflect the correct information.

Evaluation: This change to UFSAR Section 9.1.4.3.1 does not create an Unreviewed Safety Question. This change only makes the UFSAR agree with the as-built conditions of the equipment. Although the margin is actually less between what was previously stated in the UFSAR, and the correct values for the motor and load brakes; the brakes are rated for twice the margin needed for safe handling of fuel.

This change to UFSAR Section 9.1.4.3.1 will reflect the correct information for the fuel hoist motor and load brakes, which is 100% of rated load. The fuel hoist is rated for 4000 lbs. The wet weight of the mast plus a fuel assembly is approximately 2000 lbs., or about half the rated load of the hoist. This gives a safety margin of about 200%. No Technical Specification changes are required. UFSAR Section 9.1.4.3.1 will be revised.

131 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 9.2.2.6.2

Description: Local level indication of the component cooling drain sump does not exist. This UFSAR change modifies Section 9.2.2.6.2 to agree with design drawings by removing reference to the local level indication. Two level switches monitor the Component Cooling Drain Sump level. These level switches start and stop the two sump pumps and provide high-high and low-low level alarms.

Evaluation: There are no unreviewed safety questions associated with this UFSAR change. Removing the reference to a local level indication on the Component Cooling Drain Sump will not affect the operation, design basis or function of any plant system, structure, or component. The operation of the Component Cooling Drain Sump is not required for any accident evaluated in the UFSAR. Therefore removing this reference will have no effect on the probability or consequences of accidents evaluated in the UFSAR. No new accident scenarios will be created. No Technical Specification changes are required. A change is required for UFSAR Section 9.2.2.6.2.

87 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 9.2.2.6.2

Description: UFSAR Section 9.2.2.6.2 states that the centrifugal pump seals and the reciprocating pump stuffing box are provided with leakoffs to collect the leakage before it can leak to the atmosphere. Actually, the centrifugal pump seals and the reciprocating pump stuffing box are provided with leakoffs but are not sealed and are open to room atmosphere. The pump leakage is collected and processed by the liquid radwaste system. The centrifugal and reciprocating pump rooms are maintained at a negative pressure which prevents gases from reaching the atmosphere. The combination of these two systems (Liquid Radwaste Recycle and Auxiliary Building Ventilation) do prevent any leakage from reaching the atmosphere.

Evaluation: There are no unreviewed safety questions associated with this UFSAR change. There is no physical change to the plant. This UFSAR change is a clarification of how leakage from these pumps is prevented from entering the atmosphere. This change does not affect the probability or consequences of accidents. No Technical Specification changes are required. UFSAR Section 9.2.2.6.2 will be revised.

75 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 9.2.2.7

Description: UFSAR Section 9.2.2.7 currently states that active components of the Component Cooling System are in either continuous or frequent use during normal station operation and require no additional periodic tests. Due to revisions to the ASME Section XI Code, periodic tests are now performed on active components of the Component Cooling System. These tests ensure the reliability of these nuclear safety related components. This change modifies the UFSAR to agree with the requirements of ASME Section XI and the Catawba Nuclear Station Pump and Valve Inservice Test Program for active components.

Evaluation: These tests are performed per approved procedures of the Catawba Nuclear Station Pump and Valve Inservice Test Program. The addition of these periodic tests would not increase 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. A change is required for UFSAR Section 9.2.2.7.

190 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 9.3.2.2.3.2

Description: UFSAR Section 9.3.2.2.3.2 addresses the Post Accident Liquid Sampling System Sample Panel. The second sentence currently states that liquid collected inside the panel is routed to the Waste Evaporator Feed Tank Sump A during normal operation and to the Volume Control Tank during post accident operation. The sample panel sumps actually return liquid to different locations for each unit. This sentence will be revised to state "During normal operation, all liquids for Unit 1 are returned to the Waste Evaporator Feed Tank Sump A. For Unit 2, all liquids are returned to Waste Evaporator Feed Tank Sump B. During post accident operation, liquids for both Units may be diverted to the Volume Control Tank.

Evaluation: There are no unreviewed safety questions associated with this UFSAR change. The change is only a clarification. This clarification does not change the operation, design basis or function of any plant system, structure or component. No Technical Specification changes are required. UFSAR Section 9.3.2.2.3.2 will be revised.

88 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 9.3.4.2.3.13

Description: UFSAR Section 9.3.4.2.3.13 was changed to indicate that there is one (not two) local differential pressure indicators for each reactor coolant pump. The instrument loop for each reactor coolant filter provides local indication and an alarm for high differential pressure. The reactor coolant filters are used to collect resin fines and particulates from the letdown flow.

Evaluation: There is no unreviewed safety question associated with this UFSAR change. The letdown flow path is not required for any accidents evaluated in the UFSAR. The flowpath is isolated during a loss of coolant accident. This change will modify the UFSAR to agree with design drawings. This change will have no effect on the probability or consequences of any accident evaluated in the UFSAR. No Technical Specification changes are required. A change is required for UFSAR Section 9.3.4.2.3.13.

128 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 9.3.4.3.4

Description: UFSAR Section 9.3.4.3.4 "Hydrostatic Testing of the Reactor Coolant System" was revised to state that hydrostatic testing of the Reactor Coolant System will be per the requirements of ASME Boiler and Pressure Vessel Code (Case N-498-1). Hydrostatic testing is no longer regularly performed as a part of the In Service Testing Program. ASME Bolier and Pressure Vessel Code Case N-498-1 allows for leak testing at normal operating temperature and pressure.

Evaluation: There is no unreviewed safety question associated with this UFSAR change. No Technical Specification changes are required. UFSAR Section 9.3.4.3.4 will be revised.

234 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 9.3.5.2

Description: UFSAR Section 9.3.5.2 describes the Boron Recycle System. This change corrects information about where the waste gas is directed. The change corrects the text to agree with UFSAR Figure 9-100 (a system flow diagram). The change has no effect on the operation, design basis, or function of the system. There are no actual plant changes associated with this change.

Evaluation: There are no unreviewed safety questions associated with this change. There is no actual change to the plant. No Technical Specification changes are required. UFSAR Section 9.3.5.2 will be revised.

246 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 9.3.5.2.1.9

Description: UFSAR Section 9.3.5.2.1.9 about the Recycle Evaporator Package is being revised to change the solids-separating efficiency between the distillate and the bottoms in the evaporator from 10E5 to 10E6. This will allow the UFSAR text to agree with the Westinghouse Technical Manual.

Evaluation: This equipment is a part of the Boron Recycle System which does not perform a safety function and is not required to mitigate the consequences of an accident. There are no unreviewed safety questions associated with this UFSAR change. No Technical Specification changes are required. UFSAR Section 9.3.5.2.1.9 will be revised.

247 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 9.3.5.5.6, Conductivity and pH.

Description: UFSAR Section 9.3.5.5.6 is being revised to correct information that there was pH instrumentation supplied for the Recycle Evaporator. Since the Recycle Evaporator condensate is very pure it was decided during construction not to install the pH equipment. This change corrects a statement that implies the pH instrumentation is available to agree with UFSAR Figures 9-100, 9-101, 9-102, 9-103, 9-104 and 9-105, which show no pH instrumentation.

Evaluation: This change does not involve any change to the operation , design basis, or function of plant systems, structures, or components. There is no unreviewed safety question associated with this change. No Technical Specification changes are required. UFSAR Section 9.3.5.5.6 will be revised.

129 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 9.4.1 "Control Room Area Ventilation"

Description: UFSAR Section 9.4.1 "Control Room Area Ventilation" was changed to delete a sentence which stated that if a high chlorine concentration is detected in both outside air intakes, operators will examine control room instrumentation and reopen the least contaminated intake. This change is needed because the station does not have chlorine concentration instrumentation located in the Control Room and procedures do not direct operators to re-open an intake. This change will make the UFSAR and the as-built configuration of the plant compatible.

Evaluation: One of the design bases of the Control Room Ventilation System is to pressurize the control room to prevent the entry of dust, dirt, smoke, toxic gases and radioactivity. This change to the UFSAR does not impact this design basis. The change is being made to ensure the UFSAR correctly reflects the as-built condition of the station. The change will delete (a) references to control room instrumentation used to determine chlorine concentration and (b) references to operators re-opening an isolated intake. Currently there is no control room instrumentation that can determine chlorine concentration at the intakes and provide a read-out for the operators. This lack of instrumentation is not a safety issue since there are chlorine detectors that are remotely located and since no requirements exist to have permanent instrumentation. There are also no requirements to re-open an intake if both should close. The closure of both intakes would cause the Control Room Ventilation System (both trains) to become inoperable. The fact that the Technical Specifications provide direction on what to do if both trains of the Control Room Ventilation System become inoperable provides evidence that the NRC has already reviewed this configuration. Thus, there are no unreviewed safety questions associated with this UFSAR change. No Technical Specification changes are required. A change is required for UFSAR Section 9.4.1.

202 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 9.5.7.2.1

Description: UFSAR Section 9.5.7.2.1 discusses the Emergency Diesel Generator Engine Lube Oil System. The fourth paragraph first sentence states that the lube oil filters and strainers are also vented but into the room itself. The strainers are normally vented to the engine crankcase. The filters are vented to the room with a normally closed isolation off the top of the filters. These vents are used to ensure the filter or strainer is completely full of oil prior to placing the unit back in service.

Evaluation: There are no unreviewed safety questions associated with this UFSAR change. The change is a clarification and has no actual effect on plant systems, structures or components. This change has no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. UFSAR Section 9.5.7.2.1 will be revised.

183 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Sections 8.3.1.1.3, 8.3.1.4.1, 8.3.1.4.2, 8.3.1.4.3, 8.3.1.4.4

Description: UFSAR Sections 8.3.1.1.3, 8.3.1.4.1, 8.3.1.4.2, 8.3.1.4.3, and 8.3.1.4.4 each have expressions that state "No common failure mode exists for any design basis event...". This change will insert the word "known" before each of the statements in the referenced sections. The revised statement will read "No known common failure mode exists for any design basis event...". Catawba Nuclear Station was originally designed and has been modified and maintained with the intention of not having any common failure mode for any design basis event.

Evaluation: This is an editorial change that has no effect on the probability or consequences of accidents evaluated in the UFSAR. There are no unreviewed safety questions associated with this change. No Technical Specification changes are required. UFSAR Sections 8.3.1.1.3, 8.3.1.4.1, 8.3.1.4.2, 8.3.1.4.3, and 8.3.1.4.4 will be revised.

203 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Sections 9.3.4.2.2.2 and 9.3.4.2.3.7

Description: This change to UFSAR Sections 9.3.4.2.2.2 and 9.3.4.2.3.7 addressess a reduction in the limit of hydrogen from 25-35 cc hydrogen (at STP per kilogram of water) to greater than or equal to 15 cc hydrogen in Mode 2 and 25-50 cc hydrogen in Mode 1 (at STP per kilogram of water). Primary and Secondary Chemistry Specifications are a combination of EPRI Guidelines, NRC - UFSAR imposed specifications, Vendor recommendations, associated with various analytical instruments used at Catawba. Per EPRI PWR Primary Water Chemistry Guidelines: Revision 3; "Hydrogen is added to maintain reducing conditions in the primary coolant to minimize primary system corrosion. Oxidizing conditions would lead to increased formation and transport of corrosion products, higher radiation fields, reactivity, crud buildup on fuel, and increased corrosion of fuel rods. The computations of production rates of oxidizing species by radiolysis suggest a dissolved hydrogen concentration of approximately 15 cc/kg is sufficient to scavenge the oxidizing species under all operating conditions. Revision 0 of the guidelines set a range of 25-50 cc/kg to provide a margin against oxidizing conditions and to facilitate operational control. Revision 3 concluded the tighter limit of 25-35 cc/kg for plants with susceptible material is no longer justified and the range of 25-50 cc/kg can be applied to all plants".

Evaluation: There is no Unreviewed Safety Question associated with this UFSAR change. These UFSAR changes do not involve any changes to the operation, design basis or function of any structure, system or component (SSC). No safety or licensing issues are involved and no revisions to regulatory commitments are involved by the corrections or changes. No Technical Specification change is required. UFSAR Sections 9.3.4.2.2.2 and 9.3.4.2.3.7 will be revised.

215 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment (SLC) 16.1

Description: Selected Licensee Commitment (SLC) 16.1, Introduction was revised to make the following changes:

1. Changing the text reference to Compliance Functional Area Manual (CFAM) Section 3.8, Selected Licensee Commitments to Nuclear System Directive 221, Facility Operating License and Technical Specifications Bases Changes. Delete the statement that CFAM 3.8 is included in the SLC Manual as the Appendix.
2. Other minor editorial changes (addition of "in", change "FSAR" to "UFSAR".

Evaluation: There are no unreviewed safety questions associated with this change to the Selected Licensee Commitments. These are administrative changes. There is no effect on any plant system, structure, or component. The changes have no effect on any accidents evaluated in the UFSAR and no new accident scenarios are created. No regulatory requirements or commitments are being eliminated or reduced. No Technical Specification changes are required. Selected Licensee Commitment (SLC) 16.1 will be revised.

210 **Type:** UFSAR Change **Unit:** 2

Title: UFSAR Change to Selected Licensee Commitment (SLC) 16.5-6

Description: UFSAR Selected Licensee Commitment (SLC) 16.5-6, Reactor Coolant System Vents, identifies an operability requirement for maintaining vent paths from the reactor vessel head and pressurizer steam space. Testing requirement 16.5-6.a requires demonstrating that each vent path is operable by verifying once per 18 months that all manual isolation valves in each vent path are locked in the open position. A review of the system flow diagrams shows that there are no manually operated isolation valves in either the reactor vessel head vent path or the pressurizer steam space vent paths. Therefore this SLC testing requirement will be removed.

Evaluation: Reactor vessel head and pressurizer steam space vents provide a means to vent noncondensable gases from the reactor coolant system. The gases may inhibit core cooling during natural circulation. The piping, valves, component, and supports are nuclear safety related up to and including the second normally closed isolation valve. All closed valves have an assured power source to ensure operational readiness. Two of the four reactor head vent isolation valves are closed with power removed to reduce the probability of an uncontrolled depressurization of the reactor coolant system. The vent path is to the pressurizer relief tank which is capable of accepting steam, water, noncondensable gases and mixtures of these. There are no manual isolation valves in either vent path. There are no unreviewed safety questions associated with this change to Selected Licensee Commitment 16.5-6. This change will have no effect on any accident analyzed in the UFSAR. No systems, structures, or components are affected since there are no manual isolation valves in the vent paths from the reactor vessel head and pressurizer steam space. No Technical Specification changes are required. Selected Licensee Commitment (SLC) 16.5-6 will be revised.

214 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment (SLC) 16.7-10, Radiation Monitoring for Plant Operations

Description: Selected Licensee Commitment (SLC) 16.7-10, Radiation Monitoring for Plant Operations, was revised to clarify how many and which radiation monitors must be operable when operating the Component Cooling System. Also Remedial Action Statements D, F, and G of Table 16.7-10A were revised to reflect the bases of Technical Specifications 3.7.10, 3.7.12, 3.7.13. It was determined that operation of the Component Cooling Pump Train A with the A Train Radiation Monitor inoperable and relying on the B Train Radiation Monitor was an unacceptable configuration. This is due to the radiation monitor loss of flow alarm from Train A being blocked from reaching the Control Room when its associated A Train Component Cooling Pump Motor (s) are not in service. The SLC allows for the radation monitor to be out of service provided samples of Component Cooling are taken and analyzed for radioisotopes. The equivalent opposite train statement of the above sentences also applies.

Evaluation: There are no unreviewed safety questions associated with this change to the Selected Licensee Commitments. This change will have no effect on accidents analyzed in the UFSAR. The change will enable the correct monitoring of the Component Cooling System. No Technical Specification changes are required. Changes are required for Selected Licensee Commitment (SLC) 16.7-10 and Table 16.7-10A.

213 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment (SLC) 16.8.1, Containment Penetration Conductor Overcurrent Protective Devices

Description: Selected Licensee Commitment (SLC) 16.8.1, Containment Penetration Conductor Overcurrent Protective Devices, is being revised to make several minir corrections. The branch circuits for containment penetration overcurrent protective devices referenced in UFSAR Chapter 16 (SLC Manual) do not match the Electrical Low Voltage Breaker List and/or design documents. The branch circuits were arbitrarly assigned during development of the Low Voltage Breaker List without knowledge that the UFSAR Table could be affected. The names of the devices in the UFSAR (SLC Manual) will be changed to match the device names used in Low Voltage Breaker List and/or other design documents. Devices for which the field circuitry has been removed per plant modifications will be removed from the SLC Table.

Evaluation: Revising the names of Containment Penetration Overcurrent Protective Devices or deleting from the UFSAR devices that are no longer in use as Containment Penetration Overcurrent Protective Devices does not involve a physical change to the plant and has no effect on acccidents analyzed in the UFSAR. No new accident scenarios are created. There are no unreviewed safety questions associated with this UFSAR change. No Technical Specification changes are required. UFSAR Sectioin 16.8.1 will be revised.

212 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment (SLC) 16.9-1 and 16.9-2

Description: Selected Licensee Commitment (SLC) 16.9-1 and 16.9-2 are being changed to add an exception to the requirement to cycle valves at least once per twelve months in the water supply flowpath (SLC 16.9-1) and in the sprinkler system flowpath (SLC 16.9-2). The fire protection system containment isolation valves (1,2RF389B and 1,2RF447B) and the annulus sprinkler system isolation valve (1RF457B) are valves with electric motor operators. These valves are included in the station's IWV program. The IWV program requires cycling these valves quarterly per procedure PT/1,2/A/4200/028A. Therefore credit can be taken for cycling these valves per the IWV Program and they do not need to be cycled annually to meet SLC criteria. This SLC change will add a note to exempt these valves from the criteria of SLC 19.9-1(a)(iv) and 16.9-2(a)(ii).

Evaluation: There is no unreviewed safety question associated with this change to the Selected Licensee Commitments. The requirement is being met by another program. No Technical Specification changes are required. Selected Licensee Commitments (SLC) 16.9-1 and 16.9-2 will be revised.

176 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment (SLC) Section 16.7-9 (Standby Shutdown System)

Description: The following changes were made to Selected Licensee Commitment (SLC) Section 16.7-9 (Standby Shutdown System):

1. Remedial action b.) has been changed to reflect the term "total accumulative leakage" for the sum of the Identified Leakage, Unidentified Leakage, and Reactor Coolant Pump Seal Leakage. This change reflects the nomenclature used in Reactor Coolant System Leakage Calculation Procedures.
2. Remedial action b.) has been changed to lower the allowable value for total accumulative leakage to 20 gallons per minute (gpm) from 26 gpm. The lower value provides additional margin for flow measurement loop uncertainity and increased leakage due to temperature effects following a Standby Shutdown Facility (SSF) event. A corresponding paragraph has been added to the Bases section to describe the relationship between this 20 gpm limit and the 26 gpm capacity of the Standby Makeup Pump.
3. Testing Requirement 3 b.) has been changed to refer to the Inservice Testing Program for the Standby Makeup Pump testing requirements for pump head and flow values. This is being done to be consistent with the Technical Specification convention of referring to this IST Program Manual for speciific head and flow values. This is an editorial change and does not require further discussions per 10CFR50.59.

Evaluation: The Standby Shutdown System (SSF) is designed to mitigate the consequences of certain postulated fire, security, and Station Blackout (SBO) events by providing capabilities to maintain Hot Standby conditions by controlling and monitoring vital systems from locations external to the main control room. The SSF provides an alternate and independent means (with respect to the control room, and within 10 minutes) to maintain Hot Standby conditions following a postulated fire or security event for one or both units for a period of 72 hours, and a postulated SBO event for a 4 hour coping duration. By design, the SSF is intended to respond to those low-probability events, which render both the control room and automatic safety systems inoperable. The SSF is not designed to mitigate a design basis event (i.e. seismic event or LOCA) and is, therefore, not nuclear safety related or seismically designed (except where interfaces with existing safety related systems are used). After a Design Basis Event (DBE), the SSF is not required to perform any function.

The Standby Makeup Pump (SMP) functions as part of the SSF to provide makeup capacity to the reactor coolant system and cooling flow to the reactor coolant pump seals. The reactor coolant pump seal leak-off flow is temperature dependent (i.e., the higher the temperature, the higher the leak-off flow). During normal operation, the reactor coolant pump seals are supplied from the Centrifugal Charging Pumps (CCP) drawing from the Volume Control Tank (VCT). During the SSF event, the SMP draws from the Spent Fuel Pool (SFP). During the SSF event, there is no SFP cooling, so water injected into the reactor coolant pump seals will have a higher temperature than during normal operation. The SMP is capable of providing a makeup capacity of at least 26 gpm. The revised SLC limit of 20 gpm total accumulative leakage is based on a calculation that was performed to relate the SSF event leakage of 26 gpm at elevated reactor coolant pump seal temperatures. This more conservative limit will ensure that the Standby Makeup Pump will be capable of providing makeup and seal cooling flow equal to or greater than total leakage during the SSF Event, including margin for #1 Seal Leak-off Annunciator Setpoint, #1 Seal Leak-off Instrument Uncertainty, Check Valve Back Leakage, and increased seal leak-off flow due to heat-up of the Spent Fuel Pool. As a conservative measure, during normal power operation the total accumulative leakage (unidentified + identified + seal leak-off flows) shall be limited to 20 gpm. There are no unreviewed safety questions associated with this procedure revision. No Technical Specification changes are required. Changes are required for SLC Section 16.7-9 .

177 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment (SLC) Section 16.8-1 concerning Containment Penetration Conductor Overcurrent Protection Devices

Description: Although fuses are listed in the UFSAR Table 16.8-1A and Table 16.8-1B as backup devices for Containment Penetration Overcurrent Protective Devices (CPOPD), the Remedial Actions for Selected Licensee Commitment (SLC) Manual Section 16.8-1 does not address the use of fuses as protective devices. Containment Penetration Overcurrent Protective Devices do not always utilize redundant circuit breaker combination as described in the Remedial Actions. Other schemes such as a breaker and fuse combination or dual fuses are alsoused to meet the requirement. This revision will clarify the actions that should be taken should one of the other schemes have an inoperable overcurrent device.

Evaluation: Containment integrity ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. The overall integrated leakage rate is limited to less than or equal to 1.0 La to account for possible degradation of the containment leakage barriers between leakage tests. Containment Integrity shall be maintained for modes above Mode 5 and during accident conditions.

Electrical Penetrations are specially designed devices to allow circuits to enter or exit the containment, while maintaining a pressure barrier between inside and outside containment during normal operation and a loss of coolant accident (LOCA). Electrical penetrations meet the mechanical, electrical, and environmental requirements set forth in Regulatory Guide 1.63. Electrical penetration circuits, in which a credible fault current could exceed the ratings of the penetration, have overload circuit protectors to ensure a single overload protective device will not allow a fault current to cause mechanical damage to the penetration which could result in the loss of containment integrity. The circuit protectors are referenced in the UFSAR as Containment Penetration Overcurrent Protective Devices. Adding a clarifying statement to the UFSAR to explain that some circuits have breaker and fuse cominations or dual fuse combinations will not involve a physical change to the plant nor does it modify the existing design criteria for containment integrity or containment penetration overcurrent protective devices. There are no unreviewed safety questions associated with this UFSAR revision. No Technical Specification changes are required. UFSAR Section 16.8-1 will be revised.

159 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment (SLC) Tables 16.11-3 and 16.11-6.

Description: Table notations in SLC Tables 16.11-3 and 16.11-6 provide information regarding the requirements for Channel Operational Tests for specific radiation monitors. The Tables imply that a circuit failure and a downscale failure can be checked independently of each other. The two failures are really redundant failures and cannot be checked independently. A circuit failure will cause a downscale indication on the instrument and a downscale failure indicates that a circuit problem exists. The two failures will be combined into a single statement. The existing Channel Operational Tests, which are performed using approved procedures, are adequate to meet the intent of providing proper channel operation.

Evaluation: There is no unreviewed safety question associated with this change to the Selected Licensee Commitments. This change applies only to radiation monitoring equipment which is not nuclear safety related. This change has no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. A change is required for Selected Licensee Commitment (SLC) Tables 16.11-3 and 16.11-6.

148 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment 16.7-4, Editorial change to clarify which system channels are in a Loose Parts Monitoring System Collection Region

Description: This change clarifies which Loose Parts Monitoring System channels are in collection regions and required to be operable as described in Regulatory Guide 1.133, Rev. 1. The requirement is to have a minimum of two sensors capable of detecting acoustic disturbances at each natural collection region. These regions are reactor vessel upper and lower plenums, and each steam generator reactor coolant inlet plenum. Catawba installed three sensors on the reactor vessel upper and lower plenums, and two on each steam generator primary side. The Loose Parts Monitoring System has eight non-required sensors installed on the reactor coolant system. There is one sensor located on the secondary side of each steam generator and one sensor installed on each reactor coolant pump. These eight sensors are enhancements to the system, which will assist the operator in determining the location of a loose part. The failure of these channels will not impair the Loose Parts Monitoring System in performing its function. Therefore, they are not required to be operable for the Loose Parts Monitoring System to meet the intent of Regulatory Guide 1.133, Rev. 1

Evaluation: There is no unreviewed safety question associated with this UFSAR Change. This equipment does not have any accident initiation or accident mitigation function. No Technical Specification changes are required. A change is required for Selected Licensee Commitment Manual Section 16.7-4.

110 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment 16.7-5

Description: Selected Licensee Commitment (SLC) 16.7-5.a.1 and 16.7-5.a.2 will be changed to state that testing is required within 24 hours after each valve is opened. SLC 16.7-5 concerns Turbine Overspeed Protection. This SLC (16.7-5) requires at least one turbine overspeed protection system to be operable. The bases for SLC 16.7-5 states "This commitment is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are operable and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety-related components, equipment or structures."

SLC 16.7-5, Testing Requirements a.1/a.2, verify that all turbine steam valves (stop, intermediate stop and intercept/control valves) move through a complete cycle from the running position. The requirements are "At least once per 31/92 days while in Mode 1 and while in Mode 2 with the turbine operating." It was determined that this testing could not be performed until the turbine was operating (i.e. once the steam valves were opened). This has previously been handled procedurally using SLC 16.2.7, which gives 24 hours for completing a missed surveillance. A Regulatory Compliance position states that SLC 16.2.7 should not be entered intentionally. SLC 16.7-5 testing requirements a.1 and a.2 require testing for a particular plant condition, but the testing cannot be performed until the plant is in that condition.

As stated in SLC 16.7-5 (testing requirement a.1 and a.2), the valves listed above are to be cycled from their running positions. The running position for each valve is open, which indicates the testing on a valve cannot be started until that valve has been opened. Testing of the steam valves contains logic which prevents the test from beginning until the valves have been opened. The design of the turbine control system is such that the steam valves are not opened until a speed signal is input into the control system. As soon as the signal is input, the required valves open automatically to the correct position. This SLC change is to revise the testing requirements a.1 and a.2 for SLC 16.7-5 to say the testing is required within 24 hours after each valve is opened. This is the time frame in which the station has always tested the valves. This is also the time allowed by SLC in the instances where a missed surveillance is discovered.

Evaluation: This SLC change will revise testing requirements a.1 and a.2 for SLC 16.7-5 to state that the testing is required within 24 hours after each valve is opened. The basis for this change is that the testing cannot be performed until each valve has been opened. The overspeed protection is not being degraded by including the 24 hour time. Testing requirement a.3 requires the performance of a channel calibration every 18 months. Testing requirement a.4 requires disassembling of various valves. Testing requirements a.3 and a.4 are not changing.

Turbine overspeed protection should be considered OPERABLE once requirements a.3 and a.4 are completed. Testing requirements a.1 and a.2 of SLC 16.7-5 will be completed within 24 hours after each valve is opened. The testing frequency of 31/92 days is based upon the turbine manufacturer's (GE) recommendation. The frequencies are calculated in order to prevent missiles being generated. The probability of missiles being generated is very low. Testing the valves within 24 hours of opening will not increase the likelihood of an overspeed condition being generated.

Making this change to the SLC testing requirement will not increase the probability or consequences of accidents evaluated in the UFSAR. This change does not delete any testing requirement but clarifies the time at which the testing is to be performed. There are no unreviewed safety questions associated with this SLC change. No Technical Specification changes are required. No UFSAR changes are required.

107 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment 16.8-5.

Description: Selected Licensee Commitment (SLC) Section 16.8, Electrical Power Systems, addresses the Cathodic Protection System. The Cathodic Protection System is covered by SLC 16.8-5b. 1) and SLC 16.8-5b.2). A change in the wording of the SLC was made to more accurately describe the basis of the maintenance performed on the Cathodic Protection System. The wording in the SLC stating that the Cathodic Protection System should be tested in accordance with Manufacturer's inspection procedures is misleading. The system was developed by Duke Design Engineering, and the recommendations for maintenance were specified by the responsible design engineers at that time. These inspection requirements were documented in the System Description for the Cathodic Protection System, CNSD-0010-20, Section 10.0 Periodic Maintenance and Inspection. Procedure EP/0/B/3550/01 was written to meet the requirements of the Duke specified testing. The SLC wording implies an outside vendor program specified our maintenance. Therefore, this is an editorial change that will replace the wording "Manufacturer's inspection procedures" with "Duke Power approved inspection procedures" in SLC Section 16.8-5 to more accurately describe the basis of the testing performed on the System.

Evaluation: The SLC/UFSAR change that is described in this evaluation does not result in a change to the intent, interpretation, understanding, or underlying requirements of the technical content. These changes are editorial and are considered to be non-technical. The changes do not involve any safety or licensing issues. There is no unreviewed safety question associated with this SLC Change. No Technical Specification changes are required. A change is required UFSAR Chapter 16 (Selected Licensee Commitment Section 16.8).

108 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment 16.9-1 and 16.9-3

Description: This change to Selected Licensee Commitment (SLC) 16.9-1 and 16.9-3 will remove Testing Requirement 16.9-1(a)(vii)(1) and clarify SLC 16.9-3 to indicate that only one bank of nine CO2 cylinders is required to consider the High Pressure CO2 System operable. Testing requirement 16-9-1(a) (vii) (1) requires the verification that each automatic valve in the Interior/Exterior Fire Protection Systems flow path actuates to its correct position upon actuation of the system. This testing requirement is a carryover from the Standard Technical Specifications and is not applicable to Catawba since the Catawba Interior/Exterior Fire Protection System does not have any automatic valves in the flow path. It should be noted that SLC 16.9-1(a)(vii) will also be renumbered to 16.9-1(a)(vi) as a editorial correction. This SLC change will also add a note to the Bases Section of SLC 16.9-3 to agree with the information in the Interior/Exterior Fire Protection System Design Basis Document (CNS-1499.RF.00-0001) indicating that only one bank (Main or Reserve) of CO2 Cylinders is required for operability.

Evaluation: There are no unreviewed safety questions associated with this SLC Change. The probability of a fire event or accident is not affected by clarification of surveillance requirements. No Technical Specification changes are required. A change is required UFSAR Chapter 16 (Selected Licensee Commitment Section 16.9).

153 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment 16.9-6

Description: Selected Licensee Commitment 16.9-6, Table 16-9-3 is being changed to enhance the ALARA principles associated with smoke detector testing and maintenance by removing detectors in High Radiation Areas from the scope of the Selected Licensee Commitment as summarized below:

1. The detectors in the Unit 1 Valve Gallery (Fire Zone 45) and the Unit 2 Valve Gallery (Fire Zone 33) are located in high radiation areas. Table 16.9-3 currently lists 10 smoke detectors as "Minimum Instruments Operable" for Fire Zone 33 and 13 detectors as "Minimum Instruments Operable" for Fire Zone 45. This Selected Licensee Commitment change reduces the number of "Minimum Instruments Operable" to 9 for Fire Zone 33 and 12 for Fire Zone 45.
2. The detectors in the Evaporator Concentrate Batching Room (Fire Zone 65) and the Evaporator Concentrate Holdup Tank Room (Fire Zone 65) are located in high radiation areas. Table 16.9-3 currently lists 15 smoke detectors as "Minimum Instruments Operable" for Fire Zone 65. This SLC change reduces that number from 15 to 13.

An editorial change will be made by adding a table to the bases section that summarizes the detectors in each zone that are excluded from the scope of the Selected Licensee Commitment.

Evaluation: There is no unreviewed safety question associated with this Selected Licensee Commitment change. The rooms involved contain minimal combustible loading. A program is in place to control transient combustibles in these areas. It is unlikely that a significant fire would begin in these rooms and if a fire did start it would be unlikely to spread beyond the room. The equipment in the rooms are not required to perform a safe shutdown function. There are detectors in areas adjacent to these rooms which provide assurance that a fire in the area would be detected. No Technical Specification changes are required. A change is required UFSAR Chapter 16 (Selected Licensee Commitment Section 16.9).

237 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment 16.9-7, Boration Systems Flowpaths, 16-9-9, Boration Systems Charging Pump - Shutdown

Description: Selected Licensee Commitments SLC 16.9-7 Borated Systems Flow Paths - Shutdown, and SLC 16.9-9 Borated Systems Charging Pump -Shutdown are being changed to provide optional alternatives to the Centrifugal Charging Pump currently required as the borated water flow path in shutdown modes.

In the accident analysis for inadvertent boron dilution, the borated water injection flow required is 30 gpm of 7000 ppm boric acid, which converts to 78 gpm of water from the Refueling Water Storage Tank. The Residual Heat Removal System and Safety Injection System Pumps are adequate to provide in excess of 78 gpm from the Refueling Water Storage Tank in Mode 6 under specified conditions which preclude challenging the Low Temperature Ovepressure Protection (LTOP) safety function provided by the PORVs. In any case, the further restriction that the head be removed ensures that there is no potential cold overpressure event, and that the Low Temperature Overpressure Protection System is not required to be operable per Improved Technical Specification 3.4.12. Due to the further restriction that the Residual Heat Removal System pump may not be the pump that is being applied to meet LCO 3.9.4, the Residual Heat Removal System pump is not dedicated to decay heat removal duty. Further, when this option is used, the associated pump and flowpath from the Refueling Water Storage Tank must be verified per surveillance procedures to be operable and capable of being powered by an operable emergency power source.

Evaluation: There is no Unreviewed Safety Question associated with this change. No Technical Specification change is 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, in discussions regarding Low Temperature Overpressure Protection in sections 5.2.2, 7.6.20, and 6.3.2.5, or in the Chapter 15 Mode 6 inadvertent boron dilution accident analysis. The fission product barriers of the pellet, clad, Reactor Coolant System pressure boundary and containment are not affected. The Selected Licensee Commitments Manual (UFSAR Chapter 16) will be revised.

109 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment 16.9.5

Description: Selected Licensee Commitment 16.9.5 will be changed to clarify the fact that the Turbine Driven Auxiliary Feedwater Pump pit doors (AX217F and AX260F) are exempt from Testing Requirement 16.9.5(b). This Testing Requirement currently requires the inspection of the closing mechanism and the latches for each fire door listed in Selected Licensee Commitment Table 16.9-4. Doors AX217F and AX260F, by design, are not equipped with closure devices or latching mechanisms. This Selected Licensee Commitment change will add a note to Table 16.9-4 to indicate that these two doors are exempt from this testing requirement.

Evaluation: There are no unreviewed safety questions associated with this change to Selected Licensee Commitment 16.9.5. The design basis of the Turbine Driven Auxiliary Feedwater Pump Pit doors is such that they were intentionally designed without a closure device or latching mechanism. This design approach was used based on the need for rapid egress from the pit due to the CO2 system and the need to provide pressure relief during a CO2 discharge. No Technical Specification changes are required. A change is required to UFSAR Chapter 16 (Selected Licensee Commitment Section 16.9).

77 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment Manual Section 16.9-13

Description: This evaluation addresses a revision of the Selected Licensee Commitment (SLC) 16.9-13 Bases. SLC 16.9-13 addresses the commitment for snubbers, which was formerly found in the Catawba Technical Specification 3/4.7.8. The Technical Specification was previously transferred to the SLC with no changes. The Bases change involves the addition of the following paragraphs:

The Remedial Action (72 hour requirement) was previously a Technical Specification Action Statement and as such shall not be revised without NRC review/approval as an Unreviewed Safety Question. The original Technical Specification provided the 72 hour period rather than immediately entering the Action Statement for the supported system (as generally required per Technical Specification LCO 3.0.6). Since the SLC is unchanged from the original Technical Specification the Remedial Action is to be used instead of TS LCO 3.0.6 to provide the same 72 hour period.

The Remedial Action requires that an inoperable snubber be replaced or restored within 72 hours, and an engineering evaluation be performed per Testing Requirement 16.9-13g on the attached component. This testing requirement is applicable to snubbers that fail to meet the functional test criteria provided in 16.9-13f. An evaluation is required of each failure to determine the cause, and the potential for applicability of the failure mode to other snubbers. Likewise an evaluation is required to determine if the attached components have been adversely affected by the functional failure of the snubber. It is noted that the evaluation is only required for snubbers that are inoperable due to a failure of the snubber itself to meet the functional requirements. A snubber that is inoperable due solely to being disconnected from the supported component does not necessitate a component or system evaluation, provided that the snubber itself meets the requirements of 16.9-13f. In this case the only action required is that the snubber be completely restored within 72 hours and the cause of the disconnection determined and evaluated for generic implications.

Evaluation: This change to the bases does not involve an Unreviewed Safety Question. Neither the probability nor the consequences of an accident or the probability of an equipment malfunction is increased by the change. No new accidents or types of malfunction are created, and no margin of safety is reduced. The change involves the addition of clarification information with regard to getting NRC review/approval prior to changing the SLC and also provides a more detailed explanation of the Remedial Action requirements.

This modification involves no unreviewed safety questions. No changes to the Technical Specifications are required. Revision to the Selected Licensee Commitment Section 16.9-13 (Snubbers) is required.

76 **Type:** UFSAR Change **Unit:** 1

Title: UFSAR Change to Selected Licensee Commitment Manual Section 16.9-6 for Fire Zones 177 and 178

Description: This change to the Selected Licensee Commitment Manual will correct descriptions for fire zones 177 and 178 in Table 16.9.3 of Selected Licensee Commitment 16.9.6. The correct descriptions are: Zone 177 - Filter Bed Unit 1A and Zone 178 - Filter Bed Unit 1B. Minor Modification CE-9991 revised certain plant documentation to correctly identiify Zones 177 and 178. The minor modification failed to identify the need to revise Table 16.9.3 in Selected Licensee Commitment 16.9.6.

Evaluation: Correcting a mistake in the Selected Licensee Commitment Manual will have no effect on the probability or consequences of accidents described in the UFSAR. There are no unreviewed safety questions associated with this change. No Technical Specification changes are required. A change is required for SLC 16.9.

147 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment Table 16.11-1, 16.11-2, and 16.11-3

Description: The Turbine Building Sump Demineralizer Skid and Flow Totalizer are included in Selected Licensee Commitment Table 16.11-1, 16.11-2, and 16.11-3. This equipment has been deleted and removed from use with the implementation of Modification CN-50180, Waste Monitor Tank Building. The equipment was not removed from the Selected Licensee Commitment during the design of the Waste Monitor Tank Building.

Evaluation: There is no unreviewed safety question associated with this UFSAR Change. Enhanced waste processing, monitoring, storage and release capabilities were provided to the station per modification CN-50180. This modification added 60,000 gallons of waste storage capacity and facilities to provide improved waste treatment. This modification also provided piping which allows either unit's Turbine Building Sump to be transferred to the Waste Monitor Tank Building or the Steam Generator Drain Tanks for storage in the event of a contamination event on the Turbine Building Sump. Prior to the implementation of this mod, a vendor supplied demineralizer skid that was transported between Units 1 and 2 performed this function. The skid consisted of a header that contained valves and a flow totalizer. The skid was to be installed at connections on the Turbine Building Sump Pump discharge header. Vendor supplied demineralizers were to be placed in the Turbine Building Sump area to provide waste processing prior to release. The flow totalizer and composite sampling was to account for the activity in the annual release report. The Waste Monitor Tank Building modification provided processing capabilities (via demineralizers), sampling (Radiation Monitor 0EMF57) and release volumes (Waste Monitor Tank Building Flow Instrumentation). Therefore, the Turbine Building Sump equipment was made obsolete and no longer needed. This modification failed to remove the references to this equipment from the Technical Specifications (ITS converted the equipment to the Selected Licensee Commitments).

This equipment does not have any accident initiation or accident mitigation function. No Technical Specification changes are required. A change is required for Selected Licensee Commitment Manual Table 16.11.

106 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitments, Chapter 16, Section 16.11-7, Table 16.11-5

Description: This change involves editorial, non-technical changes to the Catawba Nuclear Station Selected Licensee Commitments (SLC) Manual. The need for these changes was discovered through the Corrective Action Program.
Two items are addressed:
1. The Vent Flow Rate Monitor (d) and 2. The Vent System Sampler Flow Rate Monitor (e). The flow rate monitor refers to the Unit Vent Stack Flow rate monitoring device. The sampler flow rate monitor refers to the flow meter on the Unit Vent Radiation Monitor (EMF35, 36, and 37) skid.
The proposed changes are as follows:
1. Table 16.11-5 (Page 1 of 4), Radioactive Gaseous Effluent Monitoring Instrumentation:
Change Instrument Item 3.d to read: "Unit Vent Stack Flow Rate Monitor"
Change Applicability Items 3 a.,3b., 3d., and 3e. to " ** "
Change Instrument Item 3e. To read: "Unit Vent Radiation Monitor Flow Meter"
Change Action Item 3e to read: "G"

2. Table 16.11-6 (Page 1 of 3 and 2 of 3), Radioactive Gaseous Effluent Monitoring Instrumentation:
Change Instrument Item 3 d. to read: "Unit Vent Stack Flow Rate Monitor"
Change Modes for which Surveillance Required Items 3a.,3b., 3c., and 3d. And 3 e. to : " ** "
Change Instrument Item 3e to read: "Unit Vent Radiation Monitor Flow Meter"

Evaluation: There are no Unreviewed Safety Questions associated with this change to the Selected Licensee Commitments. These changes do not result in a change to the intent, interpretation, understanding or underlying requirements of the technical content. These changes are considered editorial and non-technical. The changes do not have any effect on any accidents analyzed in the UFSAR. No Technical Specification Changes are required. A change is required for SLC Table 16.11.

56 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 1-11

Description: UFSAR Table 1-11 "Regulatory Guide 1.97, Rev. 2 Review" is being revised. The affected part of Table 1-11 addresses Regulatory Guide 1.97, Table 2, B-12 and C-6, for Containment Sump Water Level Narrow Range. The table states that "Four instruments monitor the containment sump water level from 0 to 18 inches corresponding to approximately 1 to 19 inches depth in the sump." The statement is incorrect because of the limitations of the capacitance instrumentation to monitor level below the 4 inch level and because the 17 inch sump level is just below the sump cover grating. For these reasons this change corrects the table to state "Four instruments monitor the containment sump water level from 4 to 17 inches." This change reflects current as-built conditions. Water in the sump is maintained above the four inch level. The Operator Aid Computer uses inputs from the Containment Floor and Equipment Sump level instrumentation to monitor level between the set points and a computer alarm is generated upon detection of leakage grater than one gallon per minute.

Evaluation: There are no unreviewed safety questions associated with this change. The Containment Floor and Equipment Sumps and their associated pumps and instrumentation are not initiators of any UFSAR Chapter 15 accident. These items are not nuclear safety related. The change will not affect the ability of this instrumentation to meet the requirements of Technical Specification 3.4.15 "Reactor Coolant System Leakage Detection Instrumentation" and Regulatory Guide 1.45 "Reactor Coolant Pressure Boundary Leakage Detection Systems". The change only reflects the actual capability of the installed instrumentation. No Technical Specification changes are required. A change is required for UFSAR Table 1-11.

228 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 5-18, Table 5-27 and Table 5-36

Description: UFSAR Chapter 5, "Equipment Description", identifies 2235 psig as the "operating pressure" for reactor coolant system components. The UFSAR information does not take into consideration the change in operating pressure due to reactor coolant loop pressure drops, reactor coolant pump head, and the elevation head of the pressurizer.

Westinghouse NSAL 99-002, "Operating Pressure Inconsistency", identified that the pressure at the reactor coolant pump discharge could be as high as 2309 psig, or 3.3% higher than the normal pressurizer operating pressure of 2235 psig. Thus it may be confusing to identify a "normal" reactor coolant system pressure when the pressure varies with location throughout the system piping and components. DPC-1552.08-00-0159, "RETRAN-02 Plant Simulation Model for McGuire Units 1 and 2 and Catawba Units 1 and 2", Revision 0, identifies general operation parameters applicable to the Catawba Reactor Coolant System. The RETRAN-02 Model calculates a maximum pressure of 2300 psig occurring at the reactor vessel lower plenum during normal power operation.

An operating pressure of 2235 psig is identified in UFSAR Table 5-18, "Reactor Vessel Design Parameters", Table 5-27, "Reactor Coolant Piping Design Parameters", and Table 5-36, "Reactor Coolant System Boundary Valve Design Parameters". The UFSAR does not identify an operating pressure in Table 5-23, "Reactor Coolant Pump Design Parameters", Table 5-25, "Steam Generator Design Data", Table 5-32, "Pressurizer Design Data", or Table 5-37, "Pressurizer Valve Design Data". The information presented in all these tables is similar; therefore, the UFSAR has inconsistently presented operating pressures for reactor coolant system components.

This UFSAR change will remove "operating pressure" from UFSAR Tables 5-18, 5-27, and 5-36.

Evaluation: This change request removes confusing information. It does not alter plant structures, systems, or components. There will be no change to plant procedures, system operation, or event response requirements. This change request does not remove noteworthy information from the UFSAR Sections being changed. The Catawba UFSAR identifies the pressure drop across system components in Table 5-1, "System Design And Operating Parameters". The design pressure of the reactor coolant system (2485 psig) is not being changed.

Beyond the normal operating conditions, analytical methods are used to evaluate the fluctuation in system response during Design Transients and Design Basis Events. These are defined in Section 3.9.1.1, "Design Transients", and Chapter 15, "Accident Analysis".

The analytical models for Design Transients specified a normal operating pressure of 2235 psig, without adjustment for system effects. Per Westinghouse, the maximum pressure could be 3.3% higher than the pressure specified in the analytical models. Westinghouse performed evaluations for the reactor vessel, reactor coolant piping, steam generators, and reactor coolant pumps demonstrating that consideration of this increased operating pressure would have no significant effect on the components of the reactor coolant system. Specifically, this increase in primary system operating pressure will have no significant effect on the results of stress evaluations for the reactor coolant piping system.

The analytical models for Accident Analysis adjusted for system effects related to loop pressure drops, pump head, and the elevation head. These analytical models identify that the maximum system pressure for the primary system occurred at the reactor vessel lower plenum. Therefore, the Accident Analysis models are not affected by this industry notification.

There are no unreviewed safety questions associated with this UFSAR change. The change has no effect on the probability or consequences of accidents analyzed in the UFSAR and no new accident scenarios are created. No Technical Specification changes are required. UFSAR Table 5-18, Table 5-27 and Table 5-36 will be revised.

200 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 5-37

Description: This UFSAR revision corrects the previous UFSAR Table 5-37 value for the Pressurizer Safety Valve backpressure to 500 psig from 450 psig to conform with the actual valve specification., Also, the description of this parameter is changed from "expected during discharge" to "Design" for clarification and to eliminate confusion due to the non standard description.

Evaluation: There is no associated Technical Specification change since no change is required to the surveillance criteria of Technical Specification 3.4.10 for the Pressurizer Safety Valves. The current UFSAR Analyses remain unaffected by the change in Pressurizer Safety Valve "design backpressure". The change to UFSAR Table 5-37 to show Pressurizer Safety Valve design backpressure as 500 psig has been validated by transient analysis calculations for all applicable and credible cases of Pressurizer Safety Valve and Power Operated Relief Valve opening in response to design basis accidents. It has been demonstrated that the discharge piping is large enough to limit the backpressure to less than 20% of the safety valve setpoint pressure at full flow, thus validating its sizing basis described in UFSAR Section 5.4.11.3. Since the Pressurizer Safety Valve has been shown to operate within its specified design condition for backpressure, no new failure modes are created and there is no increase in the probability of the valve malfunctioning or sticking open in response to applicable design basis accidents. Therefore, the failure modes and effects described in the corresponding UFSAR sections are not affected. There are no Unreviewed Safety Questions associated with this UFSAR change. UFSAR Table 5-37 will be revised.

163 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 6-135 (Containment Coatings)

Description: UFSAR Table 6-135 (Containment Coatings) is being revised to correct typographical errors. Also dry film thickness ranges were added and totals were deleted where required. Item 6 was deleted since this coating is not used in containment.

Evaluation: There are no unreviewed safety questions associated with this revision to Table 6-135. The changes are consistent with Coatings Specification NCMM-1167.02. These changes do not involve any nuclear safety issues, licensing issues, or regulatory commitments. These changes have no effect on the probability or consequences of accidents described in the UFSAR. No Technical Specification changes are required. UFSAR Table 6-135 will be revised.

27 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 6-74 associated with implementation of Improved Technical Specifications

Description: UFSAR Table 6-74 was changed as required for implementation of the Improved Technical Specifications. UFSAR Tables 6-77 and 6-74 are associated with each other. Changes made to Table 6-77 in previous UFSAR updates have not been reflected in Table 6-74, resulting in a number of penetrations not being included on Table 6-74. Table 6-74 was updated to reflect the changes in Table 6-77, thus clarifying the existing plant design documentation. These changes were associated with the implementation of the Improved Technical Specifications (ITS).

Several technical (non-editorial) changes have been made. The overpressure protection (bypass check valve) for seven penetrations may create the potential for bypass leak paths past the inside containment isolation valves even though the inside isolation valve is served by the Containment Penetration Valve Injection System. Table 6-74 has been changed to reflect these potential bypass leak paths. The penetrations are:

M256, Reactor Coolant Pump Seal Return, Valves NV-89A, NV-90, NV-91B
M345, Reactor Coolant Drain Tank Heat Exchanger Discharge, Valves WL-805A, WL-807B, WL-806
M221, Ventilation Unit Condensate Drain Header, Valves WL-867A, WL-869B, WL-868
M374, Containment Floor and Equip Sump Discharge, Valves WL-825A, WL-827B, and WL-321
M359, Steam Generator Drain Pump Discharge, Valves WL-A21, WL-A22, WL-A24
M321, Component Cooling from Reactor Vessel Support and Reactor Coolant Pump Coolers Vent Units, Valves KC-424B, KC-425A, KC-279
M230, Nuclear Service Water from Reactor Coolant Pump and Liquid Waste Recycle Containment Ventilation Units, Valves RN-484A, RN-487B, RN-485

Of the penetrations listed above, penetrations M221 and M230 Type C Leak Rate Test (LRT) results for the bypass check valves are included in the "combined bypass leakage" results. These seven penetrations are already included in Type C Leak Rate Test (LRT) "combined bypass leakage" results.

Since the overpressure protection valves preclude taking credit for the Containment Penetration Valve Injection System exemption in all cases, applications on Table 6-74 have been revised to remove the "X" from the applicable column. Since the Component Cooling System penetration M355 is now exempt on the basis of "Closed loop inside containment", the closed loop must be shown adequate for Post LOCA temperature. This piping was evaluated and demonstrated acceptable for all such closed loop considerations including temperature, even though the Component Cooling System flow diagram design temperature does not imply this is the case. Therefore, an "X" has been added to the column applicable to the "design temperature equals or exceeds containment design temperature for penetration M355. Therefore, all penetrations exempt from potential bypass leakage considerations are justified per the requirements of UFSAR section 6.2.4.2.1 "Containment Isolation Systems".

The Containment Valve Injection Water System is described in UFSAR 6.2.4.2.2 and shown in Figure 6-116. This system prevents leakage of containment atmosphere past certain gate valves used for containment isolation following a LOCA by injecting seal water at a pressure exceeding containment accident pressure between the two seating surfaces of the flex wedge valves. The system consists of two independent, redundant trains; one supplying gate valves that are powered by the A train diesel and the other supplying gate valves powered by the B train diesel. This separation of trains prevents the possibility of both containment isolation valves not sealing due to a single failure.

UFSAR Table 6-74, Potential Bypass Leak Paths through Containment Isolation Valves, serves to identify pathways through which containment atmosphere could bypass the filtration function performed by the Annulus Ventilation System. Since air or fluid passing through penetration piping "bypasses" the annulus, penetration isolation valves represent potential bypass leak paths for containment atmosphere unless the systems are in service following an accident or unless there are design features that preclude the escape of air containing fission products at containment accident pressure. Type C Leak Rate Testing (LRT) per Technical Specification Surveillance Requirement 3.6.1.2 is performed for those valves that represent potential bypass leak paths. Since the Containment Valve Injection Water System injects water at greater than peak containment pressure, and since this water seal system would preclude the escape of air through the inside and outside isolation valves (assuming single failure of either valve to close), it was assumed that two isolation valves served by Containment Valve Injection Water System would preclude bypass leakage. Table 6-74 previously identified those penetrations containing an inside and outside isolation valve served by the Containment Valve Injection Water System as "exempt" from bypass leakage considerations.

However, all containment penetration piping is also designed per ASME Section III, Class 2 which required ASME Code overpressure protection devices. The overpressure protection (bypass check valve) for these penetrations may create the potential for bypass leak paths past the inside containment isolation valves even though the inside isolation valve is served by the Containment Valve Injection Water System. Per UFSAR section 6.2.4.2.2, Overpressure protection is provided to relieve the pressure buildup caused by the heatup of a trapped volume of incompressible fluid between two positively closing valves (due to containment temperature transient) back into containment where an open relief path exists. Criteria 1.d under the UFSAR discussion identifies that overpressure protection must be provided for penetrations consisting of automatic closure gate valves served by the Containment Valve Injection Water System (and actuated by S, T, P, or other safety signal) both inside and outside containment. Thus, it would be more conservative to consider the bypass check valves as potential bypass leak paths since their leakage combined with a failure of the (outside) isolation valve may result in potential leakage bypassing the annulus.

To remove the potential non-conservatism the UFSAR Table 6-77 column heading "Potential Bypass Leak Path" for the seven penetrations has been changed from "No" to "Yes". The bypass check valves in two of the applications are already included in "combined bypass leakage" for the Type C. Changes to this Type C LRT procedure will be implemented so that the additional 5 bypass check valves will be included in "combined bypass leakage" test results. Including the leak rates for the additional 5 applications is a conservative measure since the reasons for excluding them cannot be positively identified.

It is noted that there are numerous, other conservative assumptions involved in the bypass leakage evaluations including inside and outside piping safety class, seismic design, and protection of the piping from pipe whip, missile and jet forces. These features are not affected by these editorial and technical discrepancies.

Evaluation: There are no unreviewed safety questions associated with this UFSAR change. These changes bring the UFSAR up to date with more conservative application of current station documents and procedures. The performance of components or systems will not be degraded by the change. No equipment used for any phase of power generation, conversion, transmission, normal shutdown cooling, fuel handling, or radwaste treatment is physically affected. Containment isolation valves are accident mitigation features and are not accident initiators. Therefore, the probability of occurrence of an accident previously evaluated in the SAR is not increased.

The performance of systems in response to an accident is not degraded. No system used to mitigate any accident is degraded. The frequency of challenges to equipment provided to mitigate accidents is not increased. The structural qualification of safety related piping has not been degraded. The post fire safe shutdown capability of the plant has not been degraded. Therefore, the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR is not increased.

Neither any fission product barrier nor any source term is adversely affected. The bypass check valves in two of the applications are already included in "combined bypass leakage" for the Type C LRT. Changes to the Type C LRT procedure will be implemented so that an additional five bypass check valves will be included in "combined bypass leakage" test results. Including the leak rates for the additional five applications is a conservative measure since the reasons for excluding them cannot be positively identified. The maximum allowable containment leakage is not affected by these changes. As a result, overall Type C leak rate testing will be more conservative, including penetrations previously not considered representing credible leak paths.

Neither any new failure modes nor any common cause failure modes are created. Rather, the changes to UFSAR Table 6-74 result in a more conservative application of Type C leak rate testing to identify seven potential containment bypass leakage pathways not previously considered to be credible (two of these pathways were already included in the "combined bypass leakage" test results). Therefore the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR are not increased.

The changes to UFSAR Table 6-74 are being made in conjunction with implementation of Catawba ITS. No Technical Specification changes are required. UFSAR Table 6-74 will be revised

26 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 6-77, Containment Isolation Valve Data

Description: UFSAR Table 6-77 underwent extensive changes after the Containment Integrity Review Report was issued in January 1996. This revision addresses errors from the Containment Integrity Review update, the addition of containment penetrations vent and drain valves, and the addition of information from previous Technical Specification Tables 3.6-1 and 3.6-2. The addition of Technical Specification Tables 3.6-1 and 3.6-2 to Table 6-77 is associated with the implementation of the Improved Technical Specifications (ITS) and is considered an editorial change. Implementation of ITS includes relocation of certain information to owner controlled documents rather than having a Technical Specification containing a list of penetrations requiring testing. The addition of the vent and drain valves introduced differences between the units so the table was broken into two sections, Unit 1 and Unit 2; however, Table 6-77 is still one table.

UFSAR Tables 6-77 and 6-74 are associated with each other. Changes made to Table 6-77 were not reflected in Table 6-74. These changes, which were identified during a review of this table, were initiated by the implementation of ITS. Table 6-74 was updated to reflected the changes in Table 6-77. This aids in clarifying the existing plant design documentation. This editorial revision will enhance the usefulness of Table 6-77 for providing guidance for plant operation, maintenance, and testing.

Column 5 "3.6.3 Applicable Condition" was added at the request of Operations. This is to aid the operators during and after implementation of ITS. ITS specification 3.6.3 deals with the operability of containment isolation valves. Prior to ITS the operators did not have to be familiar with open and closed systems in containment to determine actions and completion times. If the containment isolation was inoperable there was just one specification in the previously used Technical Specifications. ITS differentiates between open and closed systems in containment. This differentiation of systems has different completion times for different conditions. Engineering has evaluated the different penetrations associated with the systems, open or closed, and the number of isolation valves in the system, with respect to the notes associated with each condition. The conditions, "A" and "B" deal with systems and two isolations valves in the flow path; condition "C" deals with systems and one isolation valve in the flow path, the system being "closed". To aid the operator in making this determination, either an "A", "B" or "C" was put in the column. If TS 3.6.3 conditions "A", "B", or "C" did not describe the penetration it is denoted by 48 in the column, (Note 48).

The NRC has granted an exemption 10 CFR Section 50, Appendix GDC57, regarding isolation of the main steam branch lines penetrating the containment. A new general valve arrangement drawing has been created to show the Main Steam Supply to Auxiliary Equipment System valves on the main steam header "D7". The containment piping penetration valve arrangement, Figure 6-115, has a typical drawing added to it that shows the valve arrangements. D7 represents the B and C main steam headers (Table 6-77, items 91 and 92, containment penetrations M261 and M393).

For table entries 68, 69, 70, 71, 72, 73, 74, 75, 76; these penetrations represent containment purge supply and exhaust isolation valves for the lower and upper compartment, instrument room, and hydrogen purge system which are covered by a new ITS requirement. ITS Surveillance Requirement 3.6.3.1 requires that these valves be verified locked (sealed) closed every 31 days during Modes 1, 2, 3 and 4 (with certain conditional requirements relative to seat leakage only). Since these valves are verified locked (sealed) closed during the operating modes and since the dose analysis assumes they are open during the design basis fuel handling accidents, their closure times are insignificant. There are no design basis accidents resulting in containment pressurization during Modes 5, 6 and No Mode. Closure times have been changed from 5 seconds to N/A and a note of explanation has been added.

The following additional changes have been incorporated. Upon implementation of ITS at the beginning of 1999, Table 3.6-1 (Secondary Containment Bypass Leakage Paths) and 3.6-2 (Containment Isolation Valves) were removed from the previously used technical specifications and placed in UFSAR Table 6-77. This is considered an editorial change.

With the addition of vents and drain valves to Table 6-77, enough differences exist between unit valves that an expansion of the existing table into a Unit 1 and Unit 2 section was required. The table is still one long table. These are considered editorial changes since no new information is being added that is not already in the UFSAR.

The column labeled "max isolation time (sec)" is being added, from TS table 3.6-2. The times are the maximum time for the containment isolation valves to close, based on a LOCA or design accident. The information is part of the UFSAR so no new information is being added, removed, or changed. This is considered an editorial change.

The notes to Table 6-77 have been clarified as noted below:

1. The valve types and system designations were removed. These designations are part of the flow diagrams, Figures 1-22, 1-3, 1-24. This is considered an editorial change.
2. The valve arrangements in addition to Figure 6-112 are shown in Figures 6-113, 6-114, 6-115. This is considered an editorial change.
3. The relocation from old Tech Spec Tables 3.6-2, to column 13. This is considered an editorial change.
4. Notes 8 and 31 were incorporated into note 14, Table 6-77 was updated. This is considered an editorial change.
5. Note 11 was deleted since it no longer applied to the Cold Leg Accumulator valves. This is considered an editorial change.
6. Note 12 was modified to included the unit differences in the Auxiliary Feedwater System. This is considered an editorial change.
7. Note 38 was modified to include if the Containment Penetration Valve Injection System containment isolations valves failed. This is considered an editorial change.

Table 3.6-1 (Secondary Containment Bypass Leakage Paths) is incorporated into Table 6-77 for ITS.

Table 3.6-2 (Containment Isolation Valves) is a combination of UFSAR Table 6-77 columns for ITS.

The column labeled "Release Location" is being added to reflect the as built condition of the plant. These are from Tech Spec Table 3.6-1, Table 6-74 and procedure PT/1(2)/A/4200/01L, these are used to determine bypass leakage. These are considered editorial changes.

Table 6-74 items numbers were changed to reflect the same numbering that is used on Table 6-77. This is considered an editorial change.

Item 58 identified an incorrect problem investigation program reference. This reference was removed.

Evaluation: The chances of an accident are not increased as a result of these changes to UFSAR Table 6-77. These changes do not put the plant in a configuration that would increase the probability of an accident. The changes reflect the as built condition of the plant.

The information in this table is provided as an aid to the operator in order to ensure that equipment is not put in a configuration that is counter to the requirements of the Catawba Nuclear Station Technical Specifications.

The consequences of an accident as evaluated in the UFSAR are not increased by these changes. These changes do not put the plant in a configuration that would increase the consequences of an accident. The reliability of the equipment to mitigate an accident is not reduced, thus the consequences are not increased. These changes aid the operator in applying isolation measures to be in compliance with the specified "conditions" of Improved Technical Specification (ITS) 3.6.3. Therefore the accident consequences that would require containment isolation would also not increase. The assumed time to isolate for off site dose consequences is not increased as a result of this UFSAR Table change.

Equipment is put in a configuration that is in compliance with the requirements of the Catawba Nuclear Station Technical Specification; therefore, the consequences of a malfunction of equipment important to safety is not increased.

No new accident scenarios are created by this UFSAR change. All required Technical Specification provisions are met in compliance with the specified "condition" of ITS 3.6.3.

No equipment important to safety is compromised as a result of this change; therefore, the possibility of a malfunction of the equipment of a different type than evaluated in the UFSAR is not created.

The margin of safety is not decreased because all Technical Specifications directly applicable to Containment Integrity are in compliance. These changes aid the operator in applying the required containment isolation measures in compliance with specified "conditions" of ITS 3.6.3. Compliance with Technical Specification actions ensure containment integrity in the event of a single failure of any active valve.

There are no unreviewed safety questions associated with this UFSAR change. No Technical Specification changes are required. UFSAR Table 6-77 will be revised.

236 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 6-77, Unit 1 and 2 Containment Isolation Valve Data

Description: Various changes are being made to UFSAR Table 6-77, Unit 1 and 2 Containment Isolation Valve Data. The corrections or changes do not involve any changes to the operation, design basis or function of any structure, system or component. The changes to this table are shown correctly in other parts of the UFSAR or the correct information can be determined from other Sections, Figures and/or Tables in the UFSAR,

There are no physical changes or procedure changes required due to this UFSAR change.

The changes to UFSAR Table 6-77 fall into the following groups:
1) Correcting the selected valve arrangement drawing, 2) Correcting referenced UFSAR figure numbers. 3) Correcting and/or adding valve sizes and nominal line sizes. 4) Correcting whether or not seismic equipment is connected to inside and/or outside of containment. 5) Correcting valve types. 6) Correcting component numbers. 7) Correcting and/or adding notes. 8) Correcting actuator types. 9) Correcting actuation signal. 10) Correcting and/or adding normal position of valves. 11) Correcting and/or adding shutdown position of valves 12) Correcting and/or adding post accident position of valves. 13) Correcting and/or adding fail safe position. 14) Correcting valve location as to whether it is inside or outside containment. 15) Correcting governing general design criteria (GDC#). 16) Correcting type of test required (B or C). 17) Correcting whether penetration is drained for type A test or not. 18) Correcting applicable TS 3.6.3 condition. 19) Deleting superfluous information. 20) Deleting type and size of vent and/or drain valves.

Evaluation: There are no unreviewed safety questions associated with 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. There is no effect on the probability or consequences of accidents evaluated in the UFSAR. No Technical Specification changes are required. UFSAR Table 6-77 will be revised.

54 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 7-15

Description: UFSAR Table 7-15 "ESF Reponse Times" will be revised to add a note stating that the response times for valves 1(2)NW20A and 1(2)NW69B is 65/76 seconds (Normal power/Diesel Power) rather than 18/28 seconds as currently specified . This time is consistent with stroke times for other valves of the Containment Penetration Valve Injection System that do not serve as containment isolation valves. Since the design basis of the Containment Penetration Valve Injection System is to wait until the slowest valve that it serves is closed prior to injecting water, and since the design basis includes large break LOCA where the Phase A and Phase B signals are generated simultaneously, the 60 second time delay was standardized to both Phase A and Phase B closing process valves so that the slowest valve in both phases would be closed. Valves 1(2)NW20A and 1(2)NW69B receive a Phase "A" Isolation signal to open even though they are not listed as containment isolation valves in UFSAR Table 6-77 " Containment Isolation Valve Data". An operability evaluation determined that the Phase "A" Isolation response time of 18 seconds (28 seconds with Diesel Generator start and signal delays) in UFSAR Table 7-15, as referenced in Technical Specification Surveillance 3.3.2.10 , does not apply to valves 1(2)NW20A and 1(2)NW69B.

Evaluation: There are no unreviewed safety questions associated with this UFSAR change. Valve design is not being modified. The valves will continue to be tested to assure proper operation in 65/76 seconds. No Technical Specifcation revisions are required. UFSAR Table 7-15 will be revised.

130 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 7-3 "Reactor Trip System Instrumentation"

Description: UFSAR Table 7-3 "Reactor Trip System Instrumentation Maximum Time Response" is being revised. Certain response time testing criteria were identified as potentially non-conservative in Westinghouse Nuclear Safety Advisory Letter (NSAL) 98-011. The response time testing criteria determined in calculation file DPC-1552.08-00-0190, Rev 0 of March 16, 1999 was summarized and transmitted, via Engineering Instruction CNEI-0400-102, to Catawba Nuclear Station for inclusion in the UFSAR.

The new response time testing acceptance criteria are summarized below:

1. Thot and Tcold narrow range Resistance Temperature Detectors (RTD) time constants less than or equal to 6.5 seconds.
2. Thot and Tcold input to the OTDT and setpoint delays, include pressure transmitter response, less OPDT function total delays, except RTD response times, less than or equal to 1.500 seconds.
3. Pressurizer pressure input to the OTDT less than or equal to 1.500 seconds.
4. Tavg input to the OTDT and OPDT setpoints total delays, except RTD response times, less than or equal to 1.500 seconds.
5. No change to the flux imbalance input to the OTDT and OPDT setpoints delays are mandated.

Note that it is not acceptable to satisfy the sum of the RTD time response requirement and the remaining time response requirement (i.e. 8 seconds total) - each must be satisfied separately.

In addition, various editorial errors identified while processing this revision will be corrected. These editorial errors are summarized as follows:

Item 5: Maximum Time Response criteria of 8.0 (including transport time) (3) will be revised to exclude the (including transport time) portion. This was required prior to elimination of the RTD Bypass Manifold. Modifications CN10753 (Unit 1) and CN20098 (Unit 2) removed this equipment, therefore transport time is no longer applicable.

Item 6: Maximum Time Response criteria of 8.0 (including transport time) will be revised to exclude the (including transport time) portion. This was required prior to elimination of the RTD Bypass Manifold. Modifications CN10753 (Unit 1) and CN20098 (Unit 2) removed this equipment, therefore transport time is no longer applicable.

In addition, table notation (3) will be added to item 6 criteria. The Delta I component was incorporated into the OPDT function in a previous Technical Specification amendment.

This revision to Table 7-3 Items 5 and 6 of the CNS UFSAR will incorporate this criteria. Technical justification is established and documented via calculation file DPC-1552.08-00-0190, Rev 0 of March 16, 1999. This revision only distributes the previous response time allowance of 8 seconds between the individual instrumentation components of the OTDT and OPDT circuitry.

The subject editorial errors resulted from an inadvertent oversight of required UFSAR changes from implemented plant modifications.

Evaluation: The proposed revision to the Catawba UFSAR will have no effect on operation of the facility since the change does not affect the present total allowable response time of the OTDT and OPDT functions. The proposed changes will not increase the probability of an accident previously evaluated in the UFSAR. The proposed revision only constitutes redistributing the present allowable response time of the OTDT and OPDT functions among individual instrumentation components. This activity will not affect equipment operation and will not increase the probability a malfunction of equipment. The revision satisfies the previously established response time testing criteria. There will be no impact on the consequences of an accident as a result of these changes. Since this revision will only redistribute the previous allowable response time for the OTDT and OPDT functions among individual instrumentation components, this change will not create a different type of malfunction of any equipment important to safety.

There are no unreviewed safety questions associated with this revision to UFSAR Table 7-3. No Technical Specification changes are required. A change is required for UFSAR Table 7-3.

186 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 9-14

Description: UFSAR Table 9-14 (Auxiliary Feedwater Condensate Storage Tank Design Temperature) is being changed to show that the design temperature of the tank is 135 degrees F. rather than the currently listed value of 110 degrees F. The 110 degree value came from design drawing CNM-1148.00-0094 (CN-1321-16) which is from 1977. The tank was originally designed as the Filtered Water Storage Tank but was redesignated as the Auxiliary Feedwater Condensate Storage Tank with Revision 0 to the drawing which was issued on 5-5-80. A revision to calculation CNC-1148.00-06-0004 showed that the tank supports could withstand the anticipate thermal movements associated with the increase in design temperature and that slide bearing plates were not required. This tank is provided for the Auxiliary Feedwater Systems of both units as a condensate grade feedwater supply when the Auxilairy Feedwater System is not required for operability. The tank is not nuclear safety related and its availability is not assured during an accident.

Evaluation: This change has no effect on the probability or consequences of accidents analyzed in the UFSAR. No additional accidents scenarios are created by this change. There are no unreviewed safety questions associated with this UFSAR change. No Technical Specification changes are required. UFSAR Table 9-14 will be revised.

68 Type: UFSAR Change **Unit:** 1

Title: UFSAR Change to Table 9-2 and Selected Licensee Commitment 16.9-12 associated with the C1C12 Reload Safety Evaluation

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 impact 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 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 calculation 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. Table 9-2 was changed to indicate that certain information previously given in the Table is now located in the Core Operating Limits Report (COLR).

A safety evaluation of the Selected Licensee Commitment (SLC) 16.9-12 Bases revision to the boric acid tank additional margin is also performed in calculation CNC-1552.08-00-0295. The revision is required to ensure sufficient borated water exists in the Boric Acis Tank (BAT) to borate to 1.3% shutdown in going from HFP to 200 degrees F.

Evaluation: The C1C12 core reload is similar to past cycle core designs, with a design generated using NRC approved methods. No Technical Specification changes specifically related to the operation of the new C1C12 core are required. There are no unreviewed safety questions associated with the C1C12 core reload. There are no unreviewed safety questions concerning the SLC 16.9-12 Bases revision. No Technical Specification changes are required. A UFSAR change is required for Table 9-2.

256 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 9-22 (page 3 of 7)

Description: UFSAR Table 9-22 page 3 of 7 is being revised to make the following changes to Seal Water Heat Exchanger parameters:

1. Design Flow in lb/hr for Normal Condition is changed from 66,000 to 48,400.
2. Inlet Temperature in Degrees F. for Normal Condition is changed from 139 to 155.9.
3. Outlet temperature in Degrees F. for Design Condition is changed from 118 to 121.

Another column called "Alt 1" is introduced with a Heat Transfer Rate of 1.604 x 10E6, a design flow of 66,000 lb/hr and an outlet temperature of 139 degrees F. The following footnote is added which refers to the Alt 1 column: "includes max. NC Pump #1 Seal Leakage of 48 gpm.

This corrects a discrepancy between the Heat Exchanger Manufacturer's data and the data previously listed in the UFSAR.

Evaluation: The Seal Water Heat Exchanger is on the non-essential Component Cooling Water System Header and does not perform a safety function. The incorrect data is not used as inputs for Chemical and Volume Control System operation or in safe shutdown analysis. Therefore the probability or consequences for accidents analyzed in the UFSAR is not increased. Correction of this information does not create an unreviewed safety question. No Technical Specification changes are required. UFSAR Table 9-22 will be revised.

115 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 9-22, Boric Acid Filter Design Pressure Change

Description: UFSAR Table 9-22, "CVCS Principal Component Data Summary" was revised to change the design pressure of the Boric Acid Filter from 200 psig to 300 psig. This is a conservative change . The change did not cause any other changes to the design parameters (such as flow, particle retention, or material) for the Boric Acid Filter.

Evaluation: The Boric Acid Filter is not a part of any accident scenario. The component design pressure is greater than the system design requirements as given on the flow diagram. There is no unreviewed safety question associated with this change. No Technical Specification change is required. UFSAR Table 9-22 will be revised to incorporate this change.

80 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 9-22, Seal Water Injection Filter Design Temperature Change

Description: UFSAR Table 9-22 was revised to change the design temperature of the Seal Water Injection Filter from 200 degrees F. to 250 degrees F. This is a conservative change. This change does not cause any other changes to the design parameters (flow, particle retention time, or material) for the Seal Water Injection Filter. The Seal Water Injection Filter is not part of any accident scenario. The component's design temperature is equal to the system design requirement given on flow drawings.

Evaluation: Changing the design temperature of this component does not affect the probability or consequences of accidents evaluated in the UFSAR. The design temperature is being changed to agree with that given on system flow drawings. There are no unreviewed safety questions associated with this UFSAR change. No Technical Specification changes are required. A change is required for UFSAR Table 9-22.

116 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 9-22, Seal Water Return Filter Design Pressure change

Description: UFSAR Table 9-22, "CVCS Principal Component Data Summary" was revised to change the design pressure of the Seal Water Return Filter from 200 psig to 300 psig to agree with vendor documentation. This is a conservative change in that the filter vendor documentation gives the pressure higher than the pressure given in the UFSAR. The change did not cause any other changes to the design parameters (such as flow, particle retention, or material) for the Seal Water Return Filter.

Evaluation: The components design pressure is greater than the system design requirements given for that part of the system on the system flow diagram. This change to the components design pressure does not involve any change to the operation, design basis or function of any plant system, structure, or component. There is no unreviewed safety question associated with this UFSAR change. No Technical Specification changes are required. UFSAR Table 9-22 will be revised to incorporate this change.

121 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 9-23, High temperature relief line indication wrong instrument referenced.

Description: UFSAR Table 9-23 "Failure Mode and Effects Analysis Chemical and Volume Control System. Active Components - Normal Plant Operation and Load Follow", Page 8 of 42 incorrectly references instrument "NVP5160" as a failure detection method for relief valve NV14. The correct instrument that provides this function is "NCP6340."

Evaluation: This change only affects the instrument number. There is no actaul change being made to the plant. There is no unreviewed safety question associated with this change. No Technical Specification changes are required. A change will be made to UFSAR Table 9-23.

122 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR change to Table 9-23, High temperature relief line indication wrong instrument referenced.

Description: UFSAR Table 9-23 "Failure Mode and Effects Analysis Chemical and Volume Control System. Active Components - Normal Plant Operation and Load Follow", Page 40 of 42 incorrectly references instrument "NVP5540" as a failure detection method for relief valve NV236B. The correct instrument that provides this function is "NVCR5440." There are two places where this change is being made.

Evaluation: This change only affects the instrument number. There is no actaul change being made to the plant. There is no unreviewed safety question associated with this change. No Technical Specification changes are required. A change will be made to UFSAR Table 9-23.

123 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 9-23, High temperature relief line indication wrong instrument referenced.

Description: UFSAR Table 9-23 "Failure Mode and Effects Analysis Chemical and Volume Control System. Active Components - Normal Plant Operation and Load Follow", corrects the Centrifugal Charging Pump identifier given in the UFSAR. In the remarks section of Table 9-23 the Centrifugal Charging Pump is identified as Pump #1. For this statement to be correct and agree with the rest of the analysis in the Table it should be Pump 1A. The statement in the UFSAR is "Centrifugal Charging Pump #1 may be isolated by closing of manual valves in the pump's suction and discharge". This statement is true for both Charging Pumps but for this to agree with the rest of the analysis, it should be Pump 1A.

Evaluation: This change only corrects the pump identifier. There is no effect on any accident analyzed in the UFSAR. There is no unreviewed safety question associated with this UFSAR change . No Technical Specification changes are required. UFSAR Table 9-23 will be revised.

146 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 9-24, Change Design Flow of the Recycle Evaporator Condensate Demineralizer

Description: The design flow for the Recycle Evaporator Condensate Demineralizer is being corrected from 35 gpm to 75 gpm.

Evaluation: The Recycle Evaporator Condensate Demineralizer is part of the Boron Recycle System which does not perform any safety related functions and is not required to mitigate the consequences of an accident, with the exception of those portions of the system required to provide containment isolation. This change does not affect the Containment Isolation function. There is no Unreviewed Safety Question associated with this UFSAR change. No Technical Specification changes are required. A change is required for UFSAR Table 9-24.

117 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 9-24, Change design head for the Recycle Evaporator Feed Pumps.

Description: UFSAR Table 9-24, "Boron Recycle System Component Data Summary", is being revised to change the design head of the Recycle Evaporator Feed Pumps from 320 feet to 302 feet. This was a typographical error in which the last two digits were transposed.

Evaluation: The Recycle Evaporator Feed Pumps are a part of the Boron Recycle System which does not perform any nuclear safety related functions and is not required for accident mitigation, with the exception of those parts of the system required to provide containment isolation. There is no unreviewed safety question associated with this UFSAR change. No Technical Specification changes are required. UFSAR Table 9-24 will be revised to incorporate this change.

79 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Table 9-24.

Description: UFSAR Table 9-24 was revised to change the design pressure of the Recycle Evaporator Condensate Demineralizer from 250 psig to 300psig. The Recycle Evaporator Condensate Demineralizer is a part of the Boron Recycle System which does not perform any nuclear related functions and is not required for accident mitigation, with the exception of those portions of the system reqired for containment isolation. The design pressure for piping connecting to the Recycle Evaporator Condensate Demineralizer is 150 psig therefore this is a conservative change.

Evaluation: The Recycle Evaporator Condensate Demineralizer does not perform any safety function and is not used to mitigate accidents. Changing the design pressure of this item will not introduce an unreviewed safety question. No Technical Specification changes are required. A revision is required for UFSAR Table 9-24.

29 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Tables 12-23, 12-24, 12-25, 12-26 and 12-28.

Description: This UFSAR change revises Tables 12-23, 12-24, 12-25, 12-26 and 12-28. These tables provide information on the applicability of the various sections of Regulatory Guide 1.52, Rev.2, to the following ventilation filter systems:

Auxiliary Building Filtered Ventilation Exhaust System
Control Room Area Ventilation System
Annulus Ventilation System
Fuel Handling Ventilation Exhaust System
Containment Purge Exhaust System

Evaluation: Regulatory Guide 1.52 was developed to provide guidance for the construction, testing and operation of ESF ventilation filter units used at nuclear power plants. Some of the guidance given in this document is unclear and has been controversial. To clarify how Catawba Nuclear Station complies with the guide, comparison tables were developed. The comparison tables were not clear and needed to be revised. The revisions made are all considered editorial and were made for clarification purposes only. No technical changes were made. The most significant changes are discussed below.

The Regulatory Guide refers to ANSI/ASME N509-76 and ANSI/ASME N510-75. Catawba uses later revisions of these standards and the Technical Specifications refer to these later revisions. The tables were amended to show that references to the 1976 and 1975 revisions should actually be to the 1980 versions of both N509 and N510.

The use of a grain thief to obtain carbon test samples was clarified.

The penetration values used for HEPA and absorber in-place testing were clarified. While the Regulatory Guide appears to imply that 0.05% is the only acceptance value allowed, a later Generic Letter (83-13) showed that 0.05% only applied to 99% efficient filters. Since only the Control Room Area Ventilation System filters are 99% efficient the 0.05% should only apply to that system. However, the Catawba Technical Specifications state that the Unit 1 acceptance value is 1% for the Auxiliary Building Filtered Ventilation Exhaust System, Annulus Ventilation System and Fuel Handling Ventilation Exhaust System filters. It also states that the Unit 1 and Unit 2 Containment Purge Exhaust System filters acceptance criteria is 1%. These values were noted as exceptions in the Regulatory Guide 1.52 comparison tables.

A new document, PT/0/A/4450/020 Ventilation Filter Testing Program (VFTP), was developed to further clarify filter testing requirements. References to this new document were made in the comparison tables.

The deviations from the methyl iodide penetrations given in Regulatory Guide 1.52, Table 2 are documented in an SER dated 8/23/91. Since these deviations have been previously reviewed and approved by the NRC, no unreviewed safety concerns exist.

The deviation from the 720 hour run time criteria found in Regulatory Guide 1.52, Table 2, has been previously reviewed by the NRC. This change resulted from operational experience with the Control Room Area Ventilation System and the change to 1440 hours has been incorporated into the Catawba Technical Specifications. These are no unreviewed safety concerns with this deviation from Regulatory Guide 1.52, Table 2.

The allowance of the Auxiliary Building Filtered Ventilation Exhaust System filters to exceed 30,000 cfm (i.e., design flow is 30,000 cfm + 10%) is not viewed as an exception to Regulatory Guide paragraph C-2-f since the guide states "approximately" 30,000 cfm as the maximum flowrate. This flowrate was established to ensure filters remained serviceable. The Auxiliary Building Filtered Ventilation Exhaust System filters are serviceable and no technical issues are involved with exceeding 30,000 cfm since the filters were designed for flows over 30,000 cfm.

The filter units associated with this UFSAR change are not accident initiators. Testing of these filter units will not create any situation that would increase the probability of an accident occurring. No changes were made to any methods of operation or testing of these filter systems as a result of this UFSAR change.

All equipment will function as designed after use of this procedure. There are no activities that will increase the probability of equipment malfunction. There are no significant changes from past interpretations of Regulatory Guide 1.52.

The consequences of an accident are not increased by use of these UFSAR tables. All definitions and limits have been reviewed and are consistent with past station practices and industry experience. All equipment will function as designed and no activities will be conducted that will degrade any of the ESF filter units.

The changes do not involve any field work or procedure changes.

As described earlier the filter units will continue to operate as designed and perform their safety functions. No new or different accidents are associated with filter testing.

There are no unreviewed safety questions associated with this UFSAR change. No Technical Specification changes are required. Changes are required to UFSAR Tables 12-23, 12-24, 12-25, 12-26 and 12-28.

74 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to the Selected Licensee Commitments Manual

Description: The following changes are being made to the Catawba Nuclear Station Selected Licensee Commitments Manual.

1. SLC 16.7-7 - Under TESTING REQUIREMENTS section, change the word 'lease' to 'least.'

2. SLC 16.7-8 - Under TESTING REQUIREMENTS section, change the word 'lease' to 'least.'

3. SLC 16.8-1 - Under COMMITMENT section, change the word 'show' to shown.

4. SLC 16.9-1 - Change lettering/numbering under TESTING REQUIREMENTS Subsection a. from 'vii' to 'vi' and 'viii' to 'vii.'

5. SLC 16.9-1 - Change fire pump testing frequency from 'once every 31 days on a 'STAGGERED TEST BASIS' to 'once every 10 days on a STAGGERED TEST BASIS.' This restores the test frequency to the same 31 day staggered testing frequency that was in effect prior to implementation of ITS since the definition of STAGGERED TEST BASIS changed with ITS.

6. SLC 16.13-1 - Change reference from Operations Management Procedure 2-2 (OMP 2-2) to OMP 1 -10. OMP 2-2 has been deleted and OMP 1- 10 now contains requirements for fire brigade staffing.

Evaluation: The SLC/UFSAR changes that are described in this evaluation do not result in a change to the intent, interpretation, understanding, or underlying requirements of the technical content. These changes are editorial and are considered to be non-technical. The changes do not involve any safety or licensing issues. No unreviewed safety questions are created by this change to the Selected Licensee Commitments Manual. No Technical Specification Changes are required.

85 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to the Selected Licensee Commitments Manual Sections 16.9-1 and 16.9-3

Description: This change will remove testing requirements 16.9-1(a)(vii)(1) and clarify Selected Licensee Commitment 16.9-3 to indicate only one bank of nine CO2 cylinders is required to consider the High Pressure CO2 System operable. Testing requirement16.9-1(a)(viii)(1) requires the verification that each automatic valve in the Fire Protection System flow path actuates to its correct position upon actuation of the system. This testing requirement is a carryover from the Standard Technical Specifications and is not applicable to Catawba since the Catawba Fire Protection Water System does not have any automatic valves in the flow path. This change will also add a note to the bases section of Selected Licensee Commitment 16.9-3 to agree with information in the fire protection system design basis document indicating that only one bank (main or reserve) of CO2 cylinders are required for operability

Evaluation: This change does not involve an unreviewed safety question. Accident probabilities and accident consequences will not be impacted by deleting requirements that are not applicable to the design of the Catawba Fire Protection System. No Technical Specification changes are required. No UFSAR changes are required.

242 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Changes to Chapter 6 and Chapter 15 per Calculation CNC-1552.08-00-0310.

Description: Changes are being made to UFSAR Chapter 6 (Section 6.2 and Section 6.3) and Chapter 15 per calculation CNC-1552.08-00-0310. The proposed changes to these sections and the associated tables and figures do not change the current operation, design bases or function of any structure, system, component. The changes are all either non-technical editorial changes or changes made to accurately reflect the information in supporting documentation. None of the changes constitute a physical change to the plant design, configuration, or operation. The changes do not involve any safety or licensing issues and do not revise any regulatory commitments nor do they affect any requirements specified in the Technical Specifications.

Evaluation: There is no Unreviewed Safety Question associated with these UFSAR changes. No Technical Specification change is required. Many of these changes are the result of previous 10CFR50.59 evaluations, previous NRC reviews, analyses or re-analyses performed with NRC approved methods to support changes to the plant, or non-technical editorial changes. A small number of these changes have not been reviewed by the NRC. These include addition or removal of details to reflect 1) revised analyses performed with approved methods 2) revised analyses that are based on conservative changes to the NRC approved methods, and an error correction of an incorrect statement. None of these changes increase the probability or consequences of accidents evaluated in the UFSAR. UFSAR changes will be made to UFSAR Chapter 6 and Chapter 15.