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April 1, 2003

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

SUBJECT: Duke Energy Corporation
Catawba Nuclear Station Unit 1 and Unit 2
Docket Numbers 50-413 and 50-414
2002 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 for the year 2002. This report is being submitted per the provisions of 10CFR50.59(d)(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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Catawba Nuclear Station

Units 1 and 2

2002 10CFR50.59 Report

April 1, 2003

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 the year 2002. The entries are organized by the type of activity being evaluated in the following order:

17 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-61387, Install a new roof on the ground level pad located above the electrical equipment and penetration rooms.

Description: Minor Modification CE-61387 will install a new roof on the ground level pad located above the electrical equipment and penetration rooms. The roof will be installed by contractor personnel Any old roofing will be removed. The new roof will consist of a vented base sheet, two layers of smooth bituminous membrane applied with adhesive, and one layer of granular surfaced membrane, all covered with a protective sheet and two inch pavers. The installation will include flashing and accessories. This area is within the protected area fence and the building is important to the safe operation of the station.

Evaluation: This modification does not change the function of the buildings. There will be no effect on station operation after the modification is complete. During implementation of the modification it would be possible to degrade the control room ventilation carbon filters due to exposure to solvent fumes generated in the roofing process. The filters were evaluated for worst case degradation from these solvents and it was concluded that precautions would be taken to minimize solvent contamination of the carbon beds. Technical Specification Surveillance Requirement 3.7.10 2 will be performed after completion of this modification to ensure the carbon filters were not damaged. Any possible degadation of the carbon beds would not be serious enough to compromise their ability to absorb radioactive iodine during a design basis accident. No UFSAR changes are required. No Technical Specification changes are required. A 10CFR50.59 evaluation concluded that this modification could be made without prior NRC approval.

11 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-61503 "In-Mast Sipping Modification for Unit 1 Fuel Handling Crane"

Description: Westinghouse Field Change Notices (FCN) DCPO-40520 and DDPO-40524 for Catawba Nuclear Station Units 1 and 2 are entitled "Refueling Machine In-Mast Sipping System". These FCN's cover the addition of the Westinghouse In-Mast Sipping System hardware to the existing Stearns-Roger refueling machines to enable the machines to detect leaking fuel assemblies. The hardware is added to the fuel mast of the refueling machine to permit the injection of air into the bottom of the mast. The air is collected at the top of the mast and passed through a radiation monitoring system. This safety evaluation is limited to the mechanical effects of the addition of the In-Mast Sipping System hardware to the existing Stearns-Roger reactor cavity refueling machines and any impact the in-mast sipping process may have on the fuel assembly.

The following items will be added to the mast:
1. An air collection manifold assembly is to be mounted on the top flange of the stationary mast.
2. Four two inch by twelve inch slots in the stationary mast near the top are to be covered to prevent cross-flow.
3. Covers are to be placed on each of the eighteen guide roller assemblies to prevent cross-flow.
4. Five of the roller covers will have an air line bracket to support the air supply tubing.
5. The air nozzle manifold is to be mounted at the bottom of the stationary mast to allow for the injection of air.

The refueling machine is described in Catawba UFSAR Section 9.1.4, "Fuel Handling System". A brief description of the fuel handling process is provided in Section 9.1.4.2.2. The refueling machine components are described in Section 9.1.4.2.3 and in Figure 9-14. The safety evaluation of the fuel handling systems is presented in Section 9.1.4.3. As noted in UFSAR Table 3-2, the refueling machine is not nuclear safety related. However, the refueling machine is qualified for both OBE and SSE seismic loadings.

The refueling machine is a rectilinear bridge and trolley system with a vertical mast extending down into the refueling water. The bridge spans the refueling cavity and runs on rails set into the edge of the refueling cavity. The refueling machine performs fuel handling operations in the containment building.

The refueling machine is classified as Non Nuclear Safety (NNS) equipment. The design of this NNS equipment must ensure that it will resist failures that could prevent any Safety Class equipment from performing its nuclear safety function. In the case of the refueling machine, the potential adverse condition would be improper movement and handling of the fuel.

Fuel sipping is the process of identifying leaking fuel assemblies by detecting gaseous or solid fission products that have escaped from breached irradiated fuel rods. The Westinghouse In-Mast Sipping System is a set of hardware that provides a means of performing on-line, quantitative leak testing of fuel assemblies in the refueling mast during normal fuel handling operations. The set of hardware includes mechanical

adapters on the refueling machine. In addition, the analysis equipment consists of a detection and recording module, a control module, and a pump and valve module.

In operation, the change in elevation of the fuel assembly from the in-core position to the full-up in-mast position will result in a differential external pressure. This pressure differential is sufficient to cause fission gasses to migrate out of any open defects in the fuel rod cladding. The sipping system is used to identify leaking fuel assemblies by detecting these fission gasses. In-mast sipping collects the existing gases before they are released to the containment atmosphere.

The mast modifications require that:

1. Mechanical additions or modifications to the mast assembly shall not adversely affect the original designed primary safe fuel handling function of the mast or interfere with existing plant structures such as the reactor flange, cavity seal ring, or any fuel assemblies.
2. Mechanical covers are not intended to be airtight and will not be; therefore, they will not maintain a seal, and will not contribute to water displacement.
3. The air collection manifold collects the air from the annulus area between the inner and the fixed mast.
4. The air nozzle manifold at the base of the fixed mast shall not intrude into the envelope required for safe handling of the fuel assemblies.
5. Loose parts and fasteners are not permitted on any mast modification hardware. Some means of positive capture is required.
6. Any mast modification hardware that will be submerged in the refueling cavity water during operation will be constructed of 300 series stainless steel or another reactor cavity approved material.
7. To the extent practical, all mast modification hardware shall be passive in nature. That is, the actual mast hardware shall operate without need of moving parts or power sources.

The process of in-mast sipping requires that air be delivered to the bottom of the fuel assembly by means of a stainless steel tube (0.5 inch nominal diameter) mounted to the side of the refueling machine mast. During this process the tube is filled with air. When sipping is not taking place, the tube fills with water because it is immersed in the refueling cavity water. There is no undue radiation exposure concern from this air column during the in-mast sipping because the tube is not directly over the fuel assembly in the refueling machine mast. That is, the radiation source is not aligned directly with the air column. Furthermore, the tubing has a small diameter over a large distance which serves to reduce radiation streaming (from fuel assemblies in the core directly below the mast) to a negligible level as described in the following paragraph.

Radiation shielding from the fuel assemblies can be provided by ten (10) feet of water cover above a single fuel assembly which reduces the dose rate to less than 2.5 mRem/hour. The closest that the end of the air supply tube, for in-mast sipping, approaches the core is 15 feet. If it is conservatively assumed that the dose rate at the end of the tube from 193 assemblies in the core is 2.5 mRem/hour, then the dose rate is 193 x 2.5 = 480 mRem/hour. It is assumed that the 0.5 inch diameter air supply tube is 25 feet long. Then from TID-25951 "Reactor Shielding for Nuclear Engineers." N. M. Schaeffer. Ed. 1973, the dose rate for a long cylindrical duct (where the length "Z" is much longer than the duct radius, "a") at the exit of the duct, is equal to Do a^2/Z^2 where Do is the

dose rate at the entrance of the duct. For the conservative situation described above, the dose rate from the full core at the exit of the duct would be 480 x (0.25^2/300^2) = 3 x 10E-4 mRem/hour, which is clearly negligible.

Operating Loads and Seismic Evaluation:
Westinghouse has performed a seismic evaluation of the In-Mast Sipping System. The seismic evaluation demonstrates that the refueling machine is structurally adequate to withstand the seismic and dead weight loads from the addition of the In-Mast Sipping System. This additional weight will not adversely affect the safe operation of the refueling machine. The In-Mast Sipping System includes both the hardware and the analysis equipment.

Thermal Hydraulic Analysis:
Westinghouse has performed an analysis assessing the thermal hydraulic effects on the fuel assemblies being leak tested by the in-mast testing rig in the refueling machine mast. The analysis assumed no testing will be done until a minimum of 72 hours after shutdown and the surrounding water has a temperature no higher than 140°F. Based on the analysis there will be no boiling in the fuel assembly as it is suspended in the enclosed refueling mast during the test Also, the sparging of air on the fuel assembly will not have an adverse effect during the leak test sequence. Fuel assembly integrity will be maintained during and after the in-mast sipping process. Procedure MRS-SSP-1105, titled "Field Installation Procedure for the In-Mast Sipping System," addresses Quality Control. All field work will be performed in accordance with WESBU WP-19.12, "Control of Field Activities," and WESBU WP-19.11, "Preparations for Field Service Work."
The field installation will eliminate the potential for foreign objects/loose parts. Provisions include securing all nuts and bolts with lock wire. Westinghouse activities will be in accordance with PTN's Foreign Materials Exclusion Controls Administration Procedure (0-ADM-0730) for Area 1 work locations. Also, by providing qualified personnel and procedures to perform the work, Westinghouse is committed to keeping occupational radiation exposure to a minimum during all phases and limiting the possible spread of contamination. All field work will be perfomed in accordance with WESBU WP-19.12, "Control of Field Activities," and WESBU WP-19.11, "Preparations for Field Service Work."

Criticality Safety:
The use of the In-Mast Sipping system does not present a criticality safety issue. Fuel assemblies will be raised from the lower core plate to the refueling elevation above the core outlet but will still remain immersed in borated water. The assembly will enter a mast attached to the manipulator crane mast and will be positioned such that the bottom of the fuel assembly will be about one inch above the bottom of the mast. While in this position, air will be injected from each corner of the assembly at an elevation below the bottom of the fuel. The purpose is to entrain gases that may come from the fuel and to carry these gas-air mixture samples into the testing device.

The following points illustrate that there are no criticality safety concerns:
1. Fuel handling guidelines require fuel assemblies to be maintained at a minimum separation distance during fuel handling. This procedure will not violate that guideline.
2. The addition of air into the fuel assembly will increase the void fraction in the water. An increase in void fraction will decrease the reactivity of the fuel assembly.
3. The refueling water contains sufficient boron to provide a large amount of negative

reactivity. The boron concentration of the Reactor Coolant System, the refueling canal, and the refueling cavity, shall be maintained within the limit specified in Catawba Units 1 and 2 Technical Specifications.

Evaluation: A Westinghouse evaluation per 10CFR50.59 concluded that the addition of In-Mast Sipping System hardware to the existing refueling machines for Catawba Units 1 and 2 could proceed without prior NRC approval. The margin of safety with respect to integrity of the refueling machine is provided, in part, by the safety factors included in the analysis and design of the machine and is not reduced by the addition of the In-Mast Sipping hardware. No Technical Specification changes are required. A change is required to UFSAR Figure 9-14

Addition of hardware to the existing refueling machine will not increase the probability for an accident previously evaluated in the UFSAR. The refueling machine is classified as quality related equipment and is seismically qualified. It is not nuclear safety related. The addition of In-Mast Sipping hardware as defined by the FCNs meets the original design requirements for the refueling machine The potential for an accident related to fuel handling is not changed due to the addition. The analysis equipment temporarily positioned on the refueling machine for in-mast sipping is lighter than one fuel assembly. The addition does not affect the existing accident analysis included in the UFSAR because the integrity of the refueling machine is unaffected by the additional equipment. Also, the mechanism designed to grasp and raise the fuel assemblies into the mast is unchanged by the addition of the In-Mast Sipping equipment.

The addition of hardware to the existing refueling machine will not increase the consequences of an accident previously evaluated in the UFSAR. A postulated fuel assembly drop would bound the consequences of any hypothetical accident involving the refueling machine with the addition of the In-Mast Sipping hardware. The subject modifications would alter none of the parameters considered in the analysis of a postulated fuel assembly drop. The installation or use of the In-Mast Sipping System does not affect the response of the plant to postulated accident conditions.

The addition of hardware to the existing refueling machine will not create the possibility of an accident which is different than already evaluated in the UFSAR. The addition does not alter the interface of a refueling machine with a fuel assembly or the capabilities of the refueling machine. The addition of the hardware does not result in the initiation of any new credible accident. As noted above any hypothetical accident involving the refueling machine with the addition of the In-Mast Sipping hardware is bounded by previously analyzed accidents.

The addition of hardware to the existing refueling machine will not increase the probability of a malfunction of equipment important to safety previously evaluated in the UFSAR. The modification does not create any new failure modes for the refueling machine. No other component or system connecting with the refueling machine could be adversely affected by the modification.

The addition of hardware to the existing refueling machine will not increase the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR. There is no mechanism for the addition of the In-Mast Sipping hardware to affect the integrity of the refueling machine. No other component or system connecting

with the refueling machine could be adversely affected by the modification.

The addition of hardware to the existing refueling machine will not create the possibility of a malfunction of equipment important to safety different than any already evaluated in the UFSAR. The addition does not alter the interfaces of the refueling machine with fuel assemblies. In-Mast Sipping hardware does not interface directly with any safety related equipment. No other component or system connecting with the refueling machine could be adversely affected by the modification.

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

Title: Minor Modification CE-61504, In-Mast Sipping modification for Unit 2 Fuel Handling Crane

Description: Westinghouse Field Change Notices (FCN) DCPO-40520 and DDPO-40524 for Catawba Nuclear Station Units 1 and 2 are entitled "Refueling Machine In-Mast Sipping System".. These FCN's cover the addition of the Westinghouse In-Mast Sipping System hardware to the existing Stearns-Roger refueling machines to enable the machines to detect leaking fuel assemblies. The hardware is added to the fuel mast of the refueling machine to permit the injection of air into the bottom of the mast. The air is collected at the top of the mast and passed through a radiation monitoring system. This safety evaluation is limited to the mechanical effects of the addition of the In-Mast Sipping System hardware to the existing Stearns-Roger reactor cavity refueling machines and any impact the in-mast sipping process may have on the fuel assembly.

The following items will be added to the mast:
1. An air collection manifold assembly is to be mounted on the top flange of the stationary mast.
2. Four two inch by twelve inch slots in the stationary mast near the top are to be covered to prevent cross-flow.
3. Covers are to be placed on each of the eighteen guide roller assemblies to prevent cross-flow.
4. Five of the roller covers will have an air line bracket to support the air supply tubing.
5. The air nozzle manifold is to be mounted at the bottom of the stationary mast to allow for the injection of air.

The refueling machine is described in Catawba UFSAR Section 9.1.4, "Fuel Handling System". A brief description of the fuel handling process is provided in Section 9.1.4.2.2. The refueling machine components are described in Section 9.1.4.2.3 and in Figure 9-14. The safety evaluation of the fuel handling systems is presented in Section 9.1.4.3. As noted in UFSAR Table 3-2, the refueling machine is not nuclear safety related. However, the refueling machine is qualified for both OBE and SSE seismic loadings.

The refueling machine is a rectilinear bridge and trolley system with a vertical mast extending down into the refueling water. The bridge spans the refueling cavity and runs on rails set into the edge of the refueling cavity. The refueling machine performs fuel handling operations in the containment building.

The refueling machine is classified as Non Nuclear Safety (NNS) equipment. The design of this NNS equipment must ensure that it will resist failures that could prevent any Safety Class equipment from performing its nuclear safety function. In the case of the refueling machine, the potential adverse condition would be improper movement and handling of the fuel.

Fuel sipping is the process of identifying leaking fuel assemblies by detecting gaseous or solid fission products that have escaped from breached irradiated fuel rods. The Westinghouse In-Mast Sipping System is a set of hardware that provides a means of performing on-line, quantitative leak testing of fuel assemblies in the refueling mast during normal fuel handling operations. The set of hardware includes mechanical

adapters on the refueling machine. In addition, the analysis equipment consists of a detection and recording module, a control module, and a pump and valve module.

In operation, the change in elevation of the fuel assembly from the in-core position to the full-up in-mast position will result in a differential external pressure. This pressure differential is sufficient to cause fission gasses to migrate out of any open defects in the fuel rod cladding. The sipping system is used to identify leaking fuel assemblies by detecting these fission gasses. In-mast sipping collects the existing gases before they are released to the containment atmosphere.

The mast modifications require that:

1. Mechanical additions or modifications to the mast assembly shall not adversely affect the original designed primary safe fuel handling function of the mast or interfere with existing plant structures such as the reactor flange, cavity seal ring, or any fuel assemblies.
2. Mechanical covers are not intended to be airtight and will not be; therefore, they will not maintain a seal, and will not contribute to water displacement.
3. The air collection manifold collects the air from the annulus area between the inner and the fixed mast.
4. The air nozzle manifold at the base of the fixed mast shall not intrude into the envelope required for safe handling of the fuel assemblies.
5. Loose parts and fasteners are not permitted on any mast modification hardware. Some means of positive capture is required.
6. Any mast modification hardware that will be submerged in the refueling cavity water during operation will be constructed of 300 series stainless steel or another reactor cavity approved material.
7. To the extent practical, all mast modification hardware shall be passive in nature. That is, the actual mast hardware shall operate without need of moving parts or power sources.

The process of in-mast sipping requires that air be delivered to the bottom of the fuel assembly by means of a stainless steel tube (0.5 inch nominal diameter) mounted to the side of the refueling machine mast. During this process the tube is filled with air. When sipping is not taking place, the tube fills with water because it is immersed in the refueling cavity water. There is no undue radiation exposure concern from this air column during the in-mast sipping because the tube is not directly over the fuel assembly in the refueling machine mast. That is the radiation source is not aligned directly with the air column. Furthermore, the tubing has a small diameter over a large distance which serves to reduce radiation streaming (from fuel assemblies in the core directly below the mast) to a negligible level as described in the following paragraph.

Radiation shielding from the fuel assemblies can be provided by ten (10) feet of water cover above a single fuel assembly which reduces the dose rate to less than 2.5 mRem/hour. The closest that the end of the air supply tube, for in-mast sipping, approaches the core is 15 feet. If it is conservatively assumed that the dose rate at the end of the tube from 193 assemblies in the core is 2.5 mRem/hour, then the dose rate is 193 x 2.5 = 480 mRem/hour. It is assumed that the 0.5 inch diameter air supply tube is 25 feet long. Then from TID-25951 "Reactor Shielding for Nuclear Engineers." N. M. Schaeffer. Ed. 1973, the dose rate for a long cylindrical duct (where the length "Z" is much longer than the duct radius, "a") at the exit of the duct, is equal to Do a^2/Z^2 where Do is the

dose rate at the entrance of the duct. For the conservative situation described above, the dose rate from the full core at the exit of the duct would be 480 x (0.25^2/300^2) = 3 x 10E-4 mRem/hour, which is clearly negligible.

Operating Loads and Seismic Evaluation:
Westinghouse has performed a seismic evaluation of the In-Mast Sipping System. The seismic evaluation demonstrates that the refueling machine is structurally adequate to withstand the seismic and dead weight loads from the addition of the In-Mast Sipping System. This additional weight will not adversely affect the safe operation of the refueling machine. The In-Mast Sipping System includes both the hardware and the analysis equipment.

Thermal Hydraulic Analysis·
Westinghouse has performed an analysis assessing the thermal hydraulic effects on the fuel assemblies being leak tested by the in-mast testing rig in the refueling machine mast. The analysis assumed no testing will be done until a minimum of 72 hours after shutdown and the surrounding water has a temperature no higher than 140°F. Based on the analysis there will be no boiling in the fuel assembly as it is suspended in the enclosed refueling mast during the test. Also, the sparging of air on the fuel assembly will not have an adverse effect during the leak test sequence. Fuel assembly integrity will be maintained during and after the in-mast sipping process. Procedure MRS-SSP-1105, titled "Field Installation Procedure for the In-Mast Sipping System," addresses Quality Control. All field work will be performed in accordance with WESBU WP-19.12, "Control of Field Activities," and WESBU WP-19.11, "Preparations for Field Service Work."
The field installation will eliminate the potential for foreign objects/loose parts. Provisions include securing all nuts and bolts with lock wire. Westinghouse activities will be in accordance with PTN's Foreign Materials Exclusion Controls Administration Procedure (0-ADM-0730) for Area 1 work locations. Also, by providing qualified personnel and procedures to perform the work, Westinghouse is committed to keeping occupational radiation exposure to a minimum during all phases and limiting the possible spread of contamination. All field work will be perfomed in accordance with WESBU WP-19.12, "Control of Field Activities," and WESBU WP-19.11, "Preparations for Field Service Work."

Criticality Safety:
The use of the In-Mast Sipping system does not present a criticality safety issue. Fuel assemblies will be raised from the lower core plate to the refueling elevation above the core outlet but will still remain immersed in borated water. The assembly will enter a mast attached to the manipulator crane mast and will be positioned such that the bottom of the fuel assembly will be about one inch above the bottom of the mast While in this position, air will be injected from each corner of the assembly at an elevation below the bottom of the fuel. The purpose is to entrain gases that may come from the fuel and to carry these gas-air mixture samples into the testing device.

The following points illustrate that there are no criticality safety concerns:
1. Fuel handling guidelines require fuel assemblies to be maintained at a minimum separation distance during fuel handling. This procedure will not violate that guideline.
2. The addition of air into the fuel assembly will increase the void fraction in the water. An increase in void fraction will decrease the reactivity of the fuel assembly.
3. The refueling water contains sufficient boron to provide a large amount of negative

reactivity. The boron concentration of the Reactor Coolant System, the refueling canal, and the refueling cavity, shall be maintained within the limit specified in Catawba Units 1 and 2 Technical Specifications.

Evaluation: A Westinghouse evaluation per 10CFR50.59 concluded that the addition of In-Mast Sipping System hardware to the existing refueling machines for Catawba Units 1 and 2 could proceed without prior NRC approval. The margin of safety with respect to integrity of the refueling machine is provided, in part, by the safety factors included in the analysis and design of the machine and is not reduced by the addition of the In-Mast Sipping hardware. No Technical Specification changes are required. No UFSAR changes are required for this modification because the required changes were made with the equivalent Unit 1 modification.

Addition of hardware to the existing refueling machine will not increase the probability for an accident previously evaluated in the UFSAR. The refueling machine is classified as quality related equipment and is seismically qualified. It is not nuclear safety related The addition of In-Mast Sipping hardware as defined by the FCNs meets the original design requirements for the refueling machine The potential for an accident related to fuel handling is not changed due to the addition. The analysis equipment temporarily positioned on the refueling machine for in-mast sipping is lighter than one fuel assembly. The addition does not affect the existing accident analysis included in the UFSAR because the integrity of the refueling machine is unaffected by the additional equipment. Also, the mechanism designed to grasp and raise the fuel assemblies into the mast is unchanged by the addition of the In-Mast Sipping equipment.

The addition of hardware to the existing refueling machine will not increase the consequences of an accident previously evaluated in the UFSAR. A postulated fuel assembly drop would bound the consequences of any hypothetical accident involving the refueling machine with the addition of the In-Mast Sipping hardware. The subject modifications would alter none of the parameters considered in the analysis of a postulated fuel assembly drop. The installation or use of the In-Mast Sipping System does not affect the response of the plant to postulated accident conditions.

The addition of hardware to the existing refueling machine will not create the possibility of an accident which is different than already evaluated in the UFSAR. The addition does not alter the interface of a refueling machine with a fuel assembly or the capabilities of the refueling machine. The addition of the hardware does not result in the initiation of any new credible accident. As noted above any hypothetical accident involving the refueling machine with the addition of the In-Mast Sipping hardware is bounded by previously analyzed accidents.

The addition of hardware to the existing refueling machine will not increase the probability of a malfunction of equipment important to safety previously evaluated in the UFSAR. The modification does not create any new failure modes for the refueling machine. No other component or system connecting with the refueling machine could be adversely affected by the modification.

The addition of hardware to the existing refueling machine will not increase the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR. There is no mechanism for the addition of the In-Mast Sipping hardware to

affect the integrity of the refueling machine. No other component or system connecting with the refueling machine could be adversely affected by the modification.

The addition of hardware to the existing refueling machine will not create the possibility of a malfunction of equipment important to safety different than any already evaluated in the UFSAR. The addition does not alter the interfaces of the refueling machine with fuel assemblies. In-Mast Sipping hardware does not interface directly with any safety related equipment. No other component or system connecting with the refueling machine could be adversely affected by the modification.

24 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-61594 (Revision 1), Add Switches in Auxiliary Safeguards Cabinets to Bypass P-12 Interlock in Mode 4 for Extended Cooldown on Condenser Steam Dump Valves

Description: Minor Modification CE-61594 provides the method for bypass of the P-12 interlock and provides a method to use additional (Banks 2 and 3) 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

The changes effected by this modification are:

1. Install a two position key operated switch in Auxiliary Safeguards Cabinets 1AUXSFGA and 1AUXSFGB to allow the ON/BYPASS and BYP INTLK functions of the control board switch to bypass all three banks of condenser dumps, not just Bank 1. This will be a two position key lock switch. In one position, the circuitry acts exactly like is does now. In the other position, the circuitry applies the bypass function to all three banks of condenser dump valve. However, the new switch does not actually perform the bypass function. The only way to do this is through the control board switch. The new switch merely allows the bypass function to apply to all three banks. This also means the new switch could be in the enable position and have no effect on the dump valves provided the control board switch is never used to initiate the bypass function.

2. The placement of a new status light will indicate when the new switches are enabled. Wiring in various cabinets is necessary to support the addition of these status lights.

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 one bank of three valves, will be provided by the additional cooldown capacity afforded by the six additional dump valves' heat removal capability at the reactor coolant system temperature at which this modification is utilized (ability to cool with all 3 banks at a reactor coolant system Tavg of 300 deg F or below).

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.

Both the above items can be exacerbated by a failure of the steam dump controller (or other component in the steam dump system) to maximum output. This failure is possible prior to this modification but the effects are different with the P-12 interlock bypassed on the other two banks of valves. All nine valves could fail open due to a failure of the steam dump controller to maximum output

Resolution of Two Major Issues above:

1. Procedure OP/1/A/6100/002 has been revised to include provisions for either Mode 6 boron concentration or the boron concentration associated with the final temperature of the cooldown prior to utilizing this modification to make available all three banks of valves for cooldown. Thus, adequate shutdown margin will be maintained and return to criticality will not be possible.

2. An evaluation was performed to assess the cooldown potential following failure of the steam dump controller after utilizing this modification (Reactor Coolant System Tavg 300 deg F or below). It was determined that the Technical Specification cooldown limit of 100 deg F/hour should not be violated due to this failure alone with all nine 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 valves results in a much worse cooldown by comparison Thus, PTS events will not be exacerbated by this alternate cooldown method.

Adequate responses are available to react/mitigate undesirable failures during the extended cooldown. If all steam dump valves fail open and create an unacceptable cooldown rate, it can be terminated with the steam dump controller "BYP INTLK" switched to the "OFF/RESET" position or the Main Steam Isolation pushbuttons (train related) if needed.

UFSAR criteria discussed in section 7.1.2.1.2, ESFAS, related to the automatic clearing of blocks of protective functions per IEEE-279-1971 has been discussed. The purpose of the P-12 interlock was discussed and the proposed change was evaluated with respect to the existing design and intent of the protection afforded by this interlock. The consensus of the group was that the IEEE standard only applies to the plant operating Modes as defined by Technical Specifications. In other modes of operation administrative controls (including procedures) are adequate to ensure the protective function is restored to operable status when required. The design of this modification will provide a status light for each train to assist in applying these administrative controls.

This modification will not degrade any of the electrical control components. The quality of the modified wiring and terminations will meet approved requirements for design changes in the nuclear safety related Auxiliary Safeguards Cabinets. The condenser dump system and components are not safety related. However, the electrical controls assuring the steam dumps ability to close, including the P-12 interlock circuitry, is safety related and the interfacing redundant solenoid valves are provided with train related controls. The seismic integrity of affected components has been evaluated and is acceptable. The Safe Shutdown Capability of the plant is not degraded by the controls changes. No power

supplies are degraded by this modification.

Revision 0 of this evaluation was included in the 2002 Catawba Nuclear Station 10CFR50.59 Annual Summary Report (Dated April 1, 2002). Revision 1 of this evaluation adds an Attachment 4 which provides details of how the modification complies with IEEE 279-1971. The conclusion of the original 10CFR50.59 evaluation was not changed.

Evaluation: A failure modes and effects analysis was performed for this modification In no case does a failure prevent P-12 from closing the valves (provided the other train is functioning properly). This is within single failure design basis assumptions.

A 10CFR50.59 evaluation of this modification concluded that it could be implemented without prior approval from the NRC. No Technical Specification changes are required. UFSAR Section 10.4.4.2 will be revised to describe an alternate cooldown method using condenser steam dumps below the design temperature of the Residual Heat Removal System.

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

Title: Minor Modification CE-61610, Deletion or abandonment of electrical cables associated with removed computer equipment

Description: This modification will either delete or abandon electrical cables that are physically present in the plant. The disposition of these cables should have been addressed when the associated computer equipment was removed from the plant. Applicable plant documents will be revised to show the status of these cables.

Evaluation: This modification addresses the disposition of cables associated with computer equipment that has already been removed from the plant. These cables are not nuclear safety related and have no effect on any of the accidents analyzed in the UFSAR. No Technical Specification changes are required. No UFSAR revisions are required. A 10CFR50.59 evaluation determined that this modification could be implemented without prior NRC approval.

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

Title: Minor Modification CE-61752, Replace the existing two inch thick paver blocks on the Auxiliary Building roof with a new system to protect the roof

Description: Minor Modification CE-61752 will replace the existing two inch thick paver blocks on the Auxiliary Building ground level pads located near the Unit 1 and 2 Reactor Buildings. These items will be replaced with a double layer of rubber matting and covered with galvanized steel grating. These areas are within the protected area. The structures involved are important to the safe operation of the station. If an option of welding the grating in place is exercised, samples of the Control Room Ventilation System filter units will be taken and a laboratory test will be performed to monitor the condition of the filter beds following the exposure to fumes generated by the welding operation.

Evaluation: This modification does not affect the function of the structures involved. Station operation will not be affected after this work has been completed. The Auxiliary Building is nuclear safety related, but the roofing and grating system is not nuclear safety related. This modification will have no effect on any accident analyzed in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required.

6 **Type:** Miscellaneous Items **Unit:** 0

Title: Application of BWU-N CHF Correlation below the first mixing vane grid for Mk-BW Fuel Design for Steady-State DNB Analysis

Description: This evaluation is for the application of the NRC approved BWU-N CHF correlation below the first mixing vane grid for the Mk-BW fuel design. The BWU-N correlation will be applied in MAP development for RPS, Nominal, and Operational statepoint conditions and the associated steady-state DNB analyses

The activity does not affect the licensing bases for the transient analyses described in Chapter 15 of the Catawba Nuclear Station UFSAR. This activity only applies to the licensing basis governing analyses performed by the CM&TH group described in Chapter 4 of the Catawba Nuclear Station UFSAR.

Evaluation: The use of the NRC approved BWU-N CHF correlation below the first mixing vane grid for the Mk-BW fuel design produces more conservative and technically more appropriate MAP limits which ensure that DNB does not occur. Therefore, this activity does not result in a design basis limit for a fission product barrier being altered or exceeded.

The application of the NRC approved BWU-N correlation below the first mixing vane grid for RFA fuel design does not result in a departure from a method of evaluation described in the UFSAR due to the following:

1. The application of the NRC approved BWU-N CHF correlation below the first mixing vane produces more conservative and technically more appropriate MAP limits, which ensure that DNB does not occur.

2. The application of the NRC approved BWU-N correlation is consistent with its intended application per BAW-10199P-A, Addendum 1. Specifically, it is used to determine MAP limits at local conditions without mixing vane grids.

3. This activity does not affect the licensing bases for the transient analyses described in Chapter 15 of the UFSAR. This activity only applies to the licensing basis governing analyses performed by the CM&TH group described in Chapter 4 of the Catawba Nuclear Station UFSAR.

A 10CFR50.59 evaluation concluded that this change could be made without prior NRC approval. No Technical Specification changes are required. No UFSAR changes are required.

5 Type: Miscellaneous Items **Unit:** 0

Title: Application of BWU-N CHF Correlation below the first mixing vane grid for RFA Fuel Design for Steady-State DNB Analysis

Description: This evaluation is for the application of the NRC approved BWU-N CHF correlation below the first mixing vane grid for the RFA fuel design. The BWU-N correlation will be applied in MAP development for RPS, Nominal, and Operational statepoint conditions and the associated steady-state DNB analyses

The activity does not affect the licensing bases for the transient analyses described in Chapter 15 of the Catawba Nuclear Station UFSAR. This activity only applies to the licensing basis governing analyses performed by the CM&TH group described in Chapter 4 of the Catawba Nuclear Station UFSAR.

Evaluation: The application of the NRC approved BWU-N CHF correlation below the first mixing vane grid for the RFA fuel design produces more conservative and technically more appropriate MAP limits which ensure that DNB does not occur. Therefore, this activity does not result in a design basis limit for a fission product barrier being altered or exceeded.

The application of the NRC approved BWU-N correlation below the first mixing vane grid for RFA fuel design does not result in a departure from a method of evaluation described in the UFSAR due to the following:

1. The application of the NRC approved BWU-N CHF correlation below the first mixing vane produces more conservative and technically more appropriate MAP limits, which ensure that DNB does not occur.

2. The application of the NRC approved BWU-N correlation is consistent with its intended application per BAW-10199P-A, addendum 1. Specifically, it is used to determine MAP limits at local conditions without mixing vane grids.

3. This activity does not affect the licensing bases for the transient analyses described in Chapter 15 of the UFSAR. This activity only applies to the licensing basis governing analyses performed by the CM&TH group described in Chapter 4 of the Catawba Nuclear Station UFSAR.

A 10CFR50.59 evaluation concluded that this change could be made without prior NRC approval. No Technical Specification changes are required. No UFSAR changes are required.

9 **Type:** Miscellaneous Items **Unit:** 1

Title: Catawba Unit 1 Fuel Cycle 14 Reload Core Design/CNC-1552.08-00-0325, C1C14 Reload Safety Evaluation, Rev.0

Description: The Catawba Unit 1 C1C14 reload core was evaluated per 10CFR50.59 to ensure that no prior NRC review and approval was required. The evaluation determines if a license amendment is required for changes in the physics parameters predicted for this reload. Changes in the fuel assembly hydraulic/mechanical design were previously evaluated in CNC-1553.26-00-0254, Revision 3 "CMTH 10CFR50.59 Evaluation of the Westinghouse Robust Fuel Assembly Design, Rev 3, February 2001". UFSAR Chapter 15 analyses were updated for Catawba Unit 2 Fuel Cycle 11 (The first Catawba Robust Fuel Assembly (RFA) Core) in a separate evaluation (CNC-1552.08-00-0311 Revision 0, C2C11 10CFR50 59 Reload Safety Evaluation, March 2000). Station modifications, changes in tests, or changes in procedures during the refueling outage will be addressed in separate evaluations.

Evaluation: The core components and configuration in which they are arranged are similar to components and configuration used in previously approved cycles. The new fuel design has been explicitly analyzed in the safety analysis. The operational limits for this reload have been developed using methods and codes previously approved by the NRC and continue to reflect the limitations imposed by the safety and design/performance analyses. Rod position limits ensure that adequate shutdown margin is available at all times in the core life. Therefore, the frequency of occurrence of an accident previously analyzed in the UFSAR will not be increased.

The fuel assemblies of this reload are compatible with each other, the reactor internals, and the fuel handling equipment. The fuel assemblies nominally interact with no equipment important to safety other than the control rods. Changes to the fuel design which may impede the function of the control rods or the reactor internals have been adequately addressed in prior 10CFR50.59 evaluations. The mechanical compatibility of the reinserted assemblies and control rods has been demonstrated by past operation. Minor changes in reactor physics parameters are expected from cycle to cycle, and have no effect on the ability of any of the control or safety systems to perform their intended functions. Therefore, the predicted operating characteristics of this reload do not increase the likelihood of a malfunction of a system, structure, or component (SSC) important to safety previously evaluated in the UFSAR.

Satisfactory completion of the REDSAR checklist ensures assumptions pertaining to core design in the safety analyses are protected, thereby ensuring the consequences of an accident previously evaluated are not increased. Likewise, cycle-specific evaluations for deviations from REDSAR values performed in support of this reload also reveal its design ultimately remains bounded by the assumptions in the safety analyses. Fuel residence times, enrichments, and isotopic inventories are representative of the values assumed for the fuel handling accidents/dose calculations documented in the UFSAR. Changes in the fuel design that result in changes in the fuel mechanical behavior under normal and accident conditions have been explicitly accounted for in the design of the core. Therefore, the consequences of an accident previously evaluated in the above mentioned analyses are not increased.

The predicted operating characteristics of this reload do not require any changes to the setpoints for any equipment important to safety. Since changes to reactor physics parameters are expected from cycle to cycle, and these changes are minor, the response of any equipment important to safety will not be impaired during this cycle. The fuel performance and operational characteristics of this reload remain bounded by the UFSAR analyses. Factors which influence dose calculations and environmental consequences remain consistent with the UFSAR analyses. Plant operating limits and practices continue to reflect the UFSAR analyses and the Technical Specification requirements Therefore, the consequences of a malfunction of SSC important to safety previously evaluated in the UFSAR will not be increased.

The operating characteristics of this reload are similar to previously approved cycles. Fuel assembly residence times, power peaking, and other factors characterizing this reload are conservative with respect to the values assumed in the UFSAR or the ones defined in the Technical Specifications. Changes in the fuel design that result in changes in the fuel mechanical behavior under normal and accident conditions have been explicitly accounted for in the design of the core. Therefore, the predicted operating characteristics of this reload do not create the possibility of an accident which is different than any accident already analyzed in the UFSAR.

The operating characteristics of this reload are similar to the operating characteristics of previously approved cycles. Operation of previous cycles has been shown to be compatible with reactivity control systems and other plant systems through testing and operating history. Since there are only minor differences in reactor physics parameters and operating characteristics between this reload and previously approved cycles, this reload does not create the possibility of malfunctions of SSC important to safety different than any already evaluated in the UFSAR.

The reload safety evaluation calculation file and the REDSAR checklist have demonstrated that the predicted physics parameters associated with this reload cycle remain bounded by the UFSAR analyses. Cycle specific parameters for C1C14 have been shown to be bounded by the assumptions made in the safety analyses and all applicable acceptance criteria are met. The Core Operating Limits Report is produced from QA Condition 1 calculation files generated with NRC approved methods for the specific purpose of ensuring acceptance criteria continue to be met in all modes of operation. Since all of the acceptance criteria have been satisfied for the C1C14 reload, no design basis limit for a fission product barrier will be exceeded or altered.

The operational limits for this reload have been developed using methods and codes previously approved by the NRC and continue to reflect the limitations imposed by the safety and design/performance analyses. Satisfactory completion of the REDSAR checklist ensures assumptions pertaining to core design in the safety analyses are protected. Likewise, cycle-specific evaluations for deviations from REDSAR values performed in support of this reload were performed using currently approved methods and codes. Therefore, no methods of evaluation have been used in the safety analysis and in establishing the design bases which have not been described in the UFSAR.

A 10CFR50.59 evaluation concluded that the changes described above could be made without prior NRC approval. No Technical Specification changes are required. No UFSAR changes are required.

21 **Type:** Miscellaneous Items **Unit:** 0

Title: Compensatory Actions required for Work Orders 98506847 and 98385070 associated with replacement and repair of fire protection system valves

Description: The installation of Temporary Modification CNTM-0093 (evaluated elsewhere in this report) provides an alternate fire suppression water source in combination with the Compensatory Actions put into action during the maintenance evolution on valves 1RY-19 and 1RY-23 will supply the Fire Protection System with sufficient water in the event of a fire within the affected boundary.

Evaluation: These Compensatory actions do not require prior NRC approval because they have been established to meet the various Selected Licensee Commitment (SLC) remedial actions that address the reduced capability of the Fire Protection System and potential effects on the Nuclear Service Water System due to a large leak from the temporary piping connected to the Nuclear Service Water System non-essential header. These compensatory actions, including fire watches and the operation of the Tempoorary Modification allow the identification and suppression of a fire at its earliest stages. The capabilities of the Temporary Modification, together with these Compensatory Actions, provide fire fighting capacity, as previously evaluated in the UFSAR, as shown by the remedial actions of SLC 16.9-1 "Fire Suppression Water Systems", 16.9-2 "Spray and/or Sprinkler Systems" , 16.9-4 "Fire Hose Stations" and 16.9-23 "Fire Hydrants". The adequacy of Catawba's fire protection features during the unavailability of the Main Fire Pumps addressed in SLC 16.9-1 is accounted for by the use of fire watches in the required areas and the availability of an alternate fire suppression water supply. The compensatory actions also, through system isolation, limit the potentail flooding from the temporary components in the event of an earthquake, to an acceptable amount to protect equipment important to safety as evaluated in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required.

25 **Type:** Miscellaneous Items **Unit:** 0

Title: Install Control Room Area (CRA) blanks in support of Tracer Gas Testing

Description: This temporary modification is required to support a Tracer Gas Test of the Control Room Boundary. This testing will measure the amount of unfiltered in-leakage into the Control Room. The temporary modification will install a blank to isolate flow from the Control Room Pressurized Filter Train to the Control Room Area on both trains of the Control Room Area Ventilation System. The temporary modification will also rebalance the Control Room Area Ventilation System. The temporary modification will blank off the Control Room Pressurized Filter Train discharge to the Control Room Area and measure the flow rate through the filter train using a pitot traverse. If the flow rate is not within the range of 5,400 - 6,600 cfm with a Control Room pressure of 0.7 - 0.9 inwg, the system Manual Volume Dampers will be adjusted. After the conclusion of the Tracer Gas Test, the Control Room Area Ventilation System will be returned to the as-found configuration or the modification will be made permanent.

Evaluation: The Control Room Area Ventilation System and Control Room Area Chilled Water System combine to
1) ensure that the control room will remain 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 this system.
This function is accomplished by pressurizing the Control Room to greater than or equal to 1/8 inch water gauge with respect to all surrounding areas, filtering the outside air used for pressurization, filtering a portion of the return air from the Control Room to clean-up the Control Room environment, and by maintaining the Control Room temperature less than or equal to 90 degrees F.

This activity meets none of the 10CFR50.59 criteria that would require a license amendment. This temporary modification will install a blank on the end of the Pressurized Filter Train ductwork inside each Control Room Area Air Handling Unit inlet plenum. As-found flow rates and manual volume damper positions will be recorded. During this activity contact will be maintained with the Control Room to ensure that the identified pressure limit within the Control Room is maintained at all times. The train of the Control Room Area Ventilation System being modified is operating but logged in the Technical Specification Action Item Log. The activity will not cause an increase in the frequency or consequences of a malfunction or accident. This activity does not create the possibility of a different accident or a malfunction of a different type. This activity also does not affect any fission product barrier or any methods of evaluation. Installation of the Control Room Area blanks has previously been done with no adverse impact to the Control Room Ventilation System. These blanks were temporarily installed under Procedure TT/0/A/9300/033 and Work Orders 98432603 and 98432604.

No Technical Specification changes are required. No UFSAR changes are required. A 10CFR50.59 evaluation concluded that this temporary modification could be installed without prior NRC approval.

13 **Type:** Miscellaneous Items **Unit:** 0

Title: Installation of Different 2100 Series Westronics Chart Recorder Model Number for existing applications

Description: This evaluation addresses allowing the use of the Westronics Model 2100C chart recorder as a replacement for the Westronics Model 2100 chart recorder. The change in model number is related to a decision to purchase Westronics chart recorders under a 10CFR50 Appendix B Program instead of using a commercial dedication process as has been done in the past.

Evaluation: This change has no effect on any accident analyzed in the UFSAR. The chart recorders are for indication only and do not have any control function over equipment which could have an effect on accidents. A 10CFR50.59 evaluation determined that this change could be made without obtaining prior approval from the NRC. No Technical Specification changes are required. No UFSAR changes are required.

23 **Type:** Miscellaneous Items **Unit:** 0

Title: Selected Licensee Commitment (SLC) 16 2-10 evaluation for exception to 24 hour time limit associated with SLC 16 9-1 due to valve replacement of 1RY19 and 1RY23 under WO 98506847 and WO 98385070

Description: The installation of temporary modification CNTM-0093 provides an alternate fire suppression water source in combination with contingency measures put into action during the maintenance evolution on fire protection system valves 1RY19 and 1RY23. Valve 1RY19 is the Main Fire Pump Cross Connect Valve and valve 1RY23 is the Exterior Loop Isolation Post Indicator Valve. The modification will ensure that the Fire Protection System is supplied with sufficient water in the event of a fire within the affected boundary. SLC 16.2-10 addresses deviations from Selected Licensee Commitmments and provides that deviations be authorized for 14 days or less and have the concurrence of the Station Manager. SLC 16.9-1 addresses "Fire Suppression Water Systems".

Evaluation: A 10CFR50.59 evaluation concluded that these changes could be implemented without prior NRC approval because they have been established to meet the various SLC remedial actions that address the reduced capability of the fire protection system and potential effects on the Nuclear Service Water System due to a large leak from the temporary piping connected to the Nuclear Service Water System nonessential header. These contingency measures, including fire watches, and the operation of the temporary modification; allow the identification and suppression of a fire at its earliest stages. The capabilities of the temporary modification together with these contingency measures, provide fire-fighting capacity as previously evaluated in the UFSAR as shown by the remedial actions of Selected Licensee Commitments 16.9-1 "Fire Suppression Water Systems", 16.9-2 "Spray and/or Sprinkler Systems" , 16.9-4 "Fire Hose Stations" and 16.9-23 "Fire Hydrants". The adequacy of Catawba's fire protection features during the unavailability of the main fire pumps addressed in SLC 16.9-1 is accounted for by the use of fire watches in the required areas and the availability of an alternate fire suppression water supply. The contingency measures also, through system isolation, limit the potential floodfing from the temporary components in the event of an earthquake, to an acceptable amount to protect equipment inportant to safety as evaluated in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required.

20 **Type:** Miscellaneous Items **Unit:** 0

Title: Temporary Fire Protection for the Auxiliary Building (TM-0093)

Description: A temporary pump, relief valve, and associated piping will be installed in the Auxiliary Building to supply the Fire Protection System with sufficient water for the use of two hose stations with the possibility of supplying a limited number of sprinklers in the event of a fire. This activity is being done in support of modification CE-71426 which will replace Fire Protection System valves 1RY19 and 1RY23. These valves cannot be replaced without isolating the Main Fire Pumps and piping that otherwise would supply water to the Fire Protection System in the Auxiliary Building The Temporary Modification will use the Nuclear Service Water System Nonessential Supply Header as a water source, and will discharge water used to control pressure and cool the temporary pump into the Nuclear Service Water System Nonessential Return Header. A relief valve will be piped through a penetration in the Auxiliary Building exterior wall to discharge to Groundwater Drainage System Sump C in the Auxiliary Service Building. The pump discharge will be connected to the Auxiliary Building portion of the Fire Protection System through valve 1RFA056. Operation of the yard hydrants is outside the scope of this activity, which only deals with supplying water to the Fire Protection System inside the Auxiliary Building and the Reactor Buildings.
This safety evaluation does not address the adequacy of the Selected Licensee Commitment Remedial Actions cited, related activities that may need to occur to support implementation of modification CE-71426, nor does it provide complete justification for the various compensatory actions referenced. It only addresses the temporary modification and its use aside from these issues.

Evaluation: This modification does not require prior NRC approval because it is supported by various SLC remedial actions and compensatory actions to address the reduced capability of the Fire Protection System and potential effects on the Nuclear Service Water System due to a large leak from the temporary piping connected to the Nuclear Service Water System nonessential header. These compensatory actions, including fire watches and this Temporary Modification itself, allow the identification and fighting of a fire at its earliest stages. The capabilities of this Temporary Modification, together with these compensatory actions, provide fire-fighting capacity as previously evaluated in the UFSAR as shown by the remedial actions of Selected Licensee Commitment (SLC) 16.9-1 "Fire Suppression Water Systems", 16.9-2 "Spray and/or Sprinkler Systems" and 16.9-4 "Fire Hose Stations". The adequacy of Catawba's fire protection features during the unavailability of the main fire pumps addressed in SLC 16.9-1 is documented in the associated compensatory actions. A compensatory action limits the potential flooding from the temporary components in the event of an earthquake to an acceptable amount to protect equipment important to safety as evaluated in the UFSAR. No Technical Specification or other UFSAR document changes are required.

Technical Specification 3.7.8 deals with the Nuclear Service Water System. This Temporary Modification does not render any systems inoperable or compromised to the extent that any limitations of the Technical Specifications are involved. This Temporary Modification only utilizes the Nuclear Service Water System Nonessential Header, which is not discussed in the Technical Specifications. Also, this Header can be isolated from the Nuclear Service Water System Essential Header to which it is connected by double isolation valves at both connections to the Essential Header, and this continues to be the

case during the use of this Temporary Modification. A compensatory action (for valves 1RY19 and 1RY23) assures continued operability of the Nuclear Service Water System and protection from Auxiliary Building flooding.

Internal flooding of the Auxiliary Building is the only accident (design event) that could be credibly initiated by this Temporary Modification. The piping of the Temporary Modification will not be seismically qualified. It could break in the event of an earthquake, resulting in flooding of the Auxiliary Building from the Nuclear Service Water System Nonessential Header, which is seismically qualified and would not necessarily be isolated after an earthquake. This Temporary Modification will require the implementation of compensatory actions to station an individual near the piping of the Temporary Modification with instructions and authorization to isolate, or notify the Control Room to remotely isolate, the Temporary Modification piping from seismically qualified piping in the event that the temporary piping should break for any reason. The two pairs of isolation valves that separate the Nonessential Header from the Essential Header can be closed from the Control Room. None of the components affected by this Temporary Modification are accident initiators as described in Chapter 6 and 15 of the UFSAR. No aspects of this modification alter any Systems, Structures, or Components such that they could become accident initiators. Since the design features of this Temporary Modification and the related compensatory action ensure that this Temporary Modification cannot initiate an accident, it cannot increase the likelihood of occurrence of an accident

This Temporary Modification provides a backup source of water for the hose racks in the Auxiliary Building portion of the Fire Protection System. It does not place the Fire Protection System out of service. However, the remaining portions of the Fire Protection System made functional by this Temporary Modification will be used to fight fires via the implementation of remedial actions in the Selected Licensee Commitments, as discussed below, and compliance with the 24 hour time limit for the Fire Protection System being out of service per SLC 16.9-1.

This Temporary Modification allows connection of temporary piping to the Nuclear Service Water System Nonessential Header. It has been determined that the temporary piping does not structurally degrade the Nuclear Service Water System header, which is and will remain seismically qualified The Temporary Modification has been designed, supported, and will be controlled such that no failure of the Temporary Modification piping can compromise the ability of the safety-related portion of the Nuclear Service Water System to accomplish its design functions in the event of an earthquake or any Design Basis Event. The piping of the Temporary Modification can either be isolated from the Nuclear Service Water System Nonessential Header, or the Nuclear Service Water System Nonessential Header can be isolated from the Nuclear Service Water System Essential Header (safety-related) in the event of the failure of the temporary piping due to an earthquake. This will prevent a malfunction of the Nuclear Service Water System or any other system, structure, or component (SSC) important to safety due to uncontrolled flooding from the temporary components. A Compensatory Action (for valves 1RY19 and 1RY23) will be in effect during this Temporary Modification to station a person to isolate the piping and equipment of the Temporary Modification from the Nuclear Service Water System Nonessential Header in the event that major leakage develops due to an earthquake or any other reason. Therefore, this Temporary Modification cannot cause or increase the likelihood of occurrence of a malfunction of any safety-related SSC.

No malfunction of any System, Structure, or Component important to safety will be

allowed as a result of a tornado missile or tornado pressure transient or failure of a fire barrier because the penetration through the exterior Auxiliary Building wall will be protected by the implementation of a Compensatory Action, which will restore the integrity of this penetration as required by the remedial actions of SLC 16.9-5, "Fire Rated Assemblies", within one hour of a tornado watch/warning in York County, SC. The equipment of this Temporary Modification will be restrained from movement in the event of an earthquake such that its potential movement during an earthquake cannot cause a malfunction of any System, Structure, or Component important to nuclear safety. The temporary piping connected to the permanent plant systems is assumed to break, but this will not cause a malfunction of any System, Structure, or Component important to safety as discussed in the previous portions of this evaluation. The movement of the pump skid during an earthquake will not result in breakage of the piping connections at the Nonessential Headers because the polyethylene piping has been evaluated to be sufficiently supported/anchored such that unacceptable loads cannot be transmitted to the Nonessential Header piping.

A "fire" is a Design Event evaluated as an "accident" with respect to 10CFR50.59 criteria. The adequacy of Catawba's fire protection features during the unavailability of the main fire pumps addressed in SLC 16.9-1 is documented in the associated compensatory actions A summary of SLCs and remedial actions is provided below. This Temporary Modification does not provide the Fire Protection System with the full capability to mitigate a fire. Several SLCs are applicable:
SLC 16.9-1, "Fire Suppresion Water Systems", requires that the fire suppression pumps and associated water supply and piping system be operable. The remedial action requires that a backup Fire Suppression Water System be established if it is inoperable for more than 24 hours The modification to replace valves 1RY19 and 1RY23 is expected to take less than 24 hours, which satisfies remedial action b) without further action and no actions are planned. The Sprinkler System is discussed further in 16.9-2 below.
SLC 16.9-2, "Spray and/or Sprinkler Systems", requires that the Sprinkler System be operable. The Sprinkler System will be connected to the source of water supplied by this Temporary Modification However, it may be isolated if required during a fire to prevent sprinkler actuation from diverting water from the hose stations. Remedial actions for an inoperable Sprinkler System as required by this SLC are to establish fire watches in accordance with Table 16.9-1 of the SLC, and to establish backup fire suppression equipment for the affected area Credit for backup fire suppression equipment is being taken for the operation of the equipment of this Temporary Modification as connected to the Nuclear Service Water System by the associated compensatory action. This will allow the operation of two hose stations. These remedial actions will be in effect to support this Temporary Modification, even though some sprinklers may be available.
SLC 16.9-4, "Fire Hose Stations", requires that the fire hose stations in the Auxiliary Building and Fuel Pools be operable. The remedial actions specified in the SLC, providing operable nearby fire hoses, cannot be accomplished, since they assume that the Fire Suppression Water System is otherwise operable, and this will not be true. This Temporary Modification is designed as an acceptable alternate remedial action to provide water to all hose stations, any two of which may be in use to fight a potential fire. Standard Review Plan (SRP) 9.5.1. item C.6.c shows that only one hose station needs to be available to satisfy NRC requirements for hose stations, however as a practical matter to protect personnel and equipment, a second hose station will also be available to backup the primary hose station should there be a fire. Therefore the SLC will be satisfied by the alternate means of this Temporary Modification and the associated compensatory action

(to open valves connecting the Temporary Modification to the Nuclear Service Water System).
SLC 16.9-5 addresses the integrity of fire barrier penetrations. This Temporary Modification will use a penetration of the wall of the Auxiliary Building (a firewall). The Compensatory Action to address operation with this firewall breached has been approved and will be followed during the use of this Temporary Modification. It requires that the penetration be closed in the event of conditions related to fire barriers and tornados as remedial actions for this SLC.
The issue of a break in the polyethylene piping connected to the Nuclear Service Water System Nonessential Header and the potential for flooding of the Auxiliary Building was addressed above. Uncontrolled flooding would eventually affect equipment important to safety. The same justification is applicable for this question in that flooding from a break of the temporary piping will be limited by means of the associated compensatory action to an amount less than already evaluated in the UFSAR. "Consequences" is understood to mean "radioactive dose". There will be no increase in consequences from this activity. This ensures that the consequences of all accidents evaluated in the UFSAR will remain bounding.

This Temporary Modification connects to the Nuclear Service Water System Nonessential Header. It has been determined that the temporary piping will not structurally degrade the Nuclear Service Water System header, which remains seismically qualified piping. The Temporary Modification has been designed, supported and will be controlled such that no failure of the Temporary Modification piping can compromise the ability of the safety-related portion of the Nuclear Service Water System to accomplish its design functions in the event of an earthquake or any Design Basis Event.

The Nuclear Service Water System will remain capable of performing its design function because the effect of any failures of the Temporary Modification are confined to the Nuclear Service Water System Nonessential Header. Any significant leakage from this header will be prevented by isolating the leak from the header, or in the worst case, isolating the nonessential header from the rest of the Nuclear Service Water System which has safety functions to perform. Since this failure of the nonessential header cannot cause a malfunction of the remaining portion of the Nuclear Service Water System, which has safety functions, there are no increased consequences of any malfunction.

The penetration through the wall of the Auxiliary Building will continue to perform its design function as assured by compensatory action. There is therefore no malfunction and no consequences. No other Systems, Structures or Components important to safety can be damaged by the equipment of the Temporary Modification moving in the event of an earthquake because this movement will be prevented by restraining the movement of the temporary equipment as discussed above. No unacceptable forces will be transmitted to the Nuclear Service Water System nonessential header as a result of seismic motion of the temporary equipment, also discussed above. It will remain possible to isolate the nonessential header from the safety-related portions of the Nuclear Service Water System. Therefore, no malfunctions evaluated in the UFSAR that have consequences associated have any increase as a result of the activity being evaluated.

This Temporary Modification is not able to initiate any accidents either directly or indirectly. There are no failures that can be postulated that could create a new type of accident not already evaluated in the UFSAR provided that the compensatory actions are

successfully implemented.
The Fire Protection System will continue to perform its functions in the combating of a design basis fire with the support of this Temporary Modification and related remedial actions from the SLCs. A leak from the temporary piping will not cause the loss of the portion of the Nuclear Service Water System important to safety, nor cause the loss of any other safety-related equipment due to flooding, because the leak will be stopped either at the connection to the Nuclear Service Water System Nonessential Header or from the Control Room by isolating the nonessential header from the rest of the Nuclear Service Water System. Thus the Nuclear Service Water System will continue to provide cooling water to safety-related equipment and will not cause any malfunction in the Nuclear Service Water System itself, or the associated equipment it cools. The wall penetration will continue to perform its function because its integrity will be restored when conditions require it by the compensatory action associated with the Auxiliary Building wall penetration No malfunctions of equipment important to safety will be caused by movement of the temporary equipment during an earthquake because this equipment will be restrained. The piping not connecting to the Nuclear Service Water System nonessential header will be laying on the floor or be away from or not otherwise able to damage any nearby safety-related equipment in the event of its failure Therefore this Temporary Modification will not create any malfunction of a System, Structure, or Component important to safety nor will it create any malfunction with a different result than previously evaluated in the UFSAR.

The potential effects of this Temporary Modification on safety-related equipment in the plant have been evaluated above. These potential effects all only indirectly affect fission product barriers if unacceptable potential consequences were postulated to remain uncorrected. There are no direct effects, and the potential indirect effects through affected Systems, Structures, or Components important to safety and the Fire Protection System have been limited to be within the effects previously evaluated in the UFSAR or are acceptable with the application of the compensatory and remedial actions identified above. Therefore, this Temporary Modification can not result in any design basis limit for fission product barriers being exceeded or altered.

This Temporary Modification does not have any relationship to a method of evaluation. It provides temporary equipment which can be used to supply water to hose stations and potentially some sprinklers of the Fire Protection System.

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

Title: Nuclear Station Modification CN-11392/00, Abandon Positive Displacemet Pump No. 1

Description: This modification will abandon-in-place the Positive Displacement Pump (PDP) No. 1 and associated Chemical and Volume Control System, Nuclear Service Water System, and Component Cooling System piping and components in Unit 1. Wiring for power, instrumentation, and control will be deleted. Interfacing controls associated with PDP No. 1 and other plant equipment will be deleted as necessary. Indications in the Control Room will be deleted, and references to the pump will be removed from Operations procedures. The UFSAR will be revised to reflect the equipment status.

This modification will involve cutting and capping of piping on the suction and discharge of PDP No. 1 and on both ends of the stuffing box head tank overflow line. Also, the Component Cooling System piping to the PDP oil cooler will be cut and capped. Additionally the Nuclear Service Water System piping to the PDP fluid drive cooler will be cut and capped.

The air operated actuator on valve 1NV-476 will be removed and the valve gagged closed. The Electric Motor Operator (EMO) for valve 1NV-477 will be electrically disconnected and the EMO will remain installed on the valve. Valve 1NV-477 will be closed and can be operated with the handwheel on the EMO. Valve 1NV-481 will be pneumatically disconnected and the limit switches removed. Valve 1NV-481 will be left closed and cannot be operated unless an air supply is connected to the valve. Valve 1NV-478 will no longer need to be locked throttled and will be shown on the flow drawing as normally closed. Relief valve 1NV-305 will be gagged closed since its purpose was overpressure protection for PDP No. 1.

Instrumentation associated with valve 1NV-477 (position control on main control board) and speed control for PDP No. 1 will be deleted from Control Board 1MC10. Certain other Chemical and Volume Control System instrumentation will be abandoned or deleted. Also instrumentation associated with the abandoned Component Cooling System and Nuclear Service Water System piping will be abandoned

Evaluation: A 10CFR50.59 evaluation concluded that this change could be made without prior NRC approval. The PDP No. 1 is not an accident initiator. The pump was removed from service by a previous modification. No Technical Specification changes are required. Changes are required to UFSAR Table 3-4, UFSAR Sections 6.3.2.5, 9.2.1.2.3, 9.2.2.2, 9.3.4.2.3.1, and 9.3.4.2.3.18, UFSAR Table 9-6, UFSAR Table 9-22, UFSAR Table 9-23, and UFSAR Table 12-19.

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

Title: Nuclear Station Modification CN-11413/00, De-energize the hydraulic solenoids on the Main Feedwater Isolation Valves

Description: The Main Feedwater Isolation Valves are designed to isolate feedwater flow to a faulted steam generator upon receipt of a feedwater isolation signal. The valves are also used to isolate feedwater to the Steam Generators during normal shutdown and startup operations These valves are designated as 1CF33, 1CF42, 1CF51, and 1CF60.

The two hydraulic solenoids in the operator of the Main Feedwater Isolation Valves are normally energized and are de-energized to close the valve. A single failure that de-energizes either of these solenoids will close the valve causing a trip of the associated unit. This modification is a reliability enhancement to steady state unit operation.

Modification CN-11413/00 will change the control circuitry for the hydraulic solenoid valves to a normally de-energized state. Currently, these valves are normally energized in a closed position, allowing the hydraulic oil pressure to be exposed to the bottom of the valve actuator by maintaining the vent flowpath isolated. Upon loss of power to these solenoid valves, the flowpath is aligned to the reservoir allowing the hydraulic oil pressure to decay, resulting in closure of the valve from the application of nitrogen pressure to the top of the actuator.

In addition, since the 10% stroke testing circuit associated with each train of main feedwater isolation valves has never been used, this testing circuit will be removed. This will result in removal of two control switches on Main Control Board 1MC2.

The new control circuit requires a new method to alert the operators to a loss of control power. The new circuit has two parallel, normally energized relays which contain two normally open contacts wired in series, per train that control the status of two parallel train related normally closed solenoid valves.

Features have been provided to indicate the presence of problems, such as a loss of power to a relay, which might otherwise go undetected and preclude operation of these Engineered Safety Features. The Operator Aid Computer will monitor the energized state of the relays. A de-energized relay closes its associated contact and nothing is inoperable but the plant is closer to a trip condition (solenoid valve opening - main feedwater isolation valve closure). The Regulatory Guide 1.47 Bypass Panel will monitor the control power to these valves.

Evaluation: This modification is considered an adverse change in that the design may be less fail safe, although still single failure proof, than the current design. The benefits of the de-energized solenoid design that would minimize the potential for a unit trip is balanced against additional complexities in the circuitry and the characteristic of having to energize the solenoids to perform a safety function. Since the change may be considered adverse, a 10CFR50.59 evaluation was performed.

Seven of the eight criteria of 10CFR50.59 were addressed using the following argument. A Feedwater Isolation causes a Reactor/Turbine Trip and turbine trips are evaluated as ANS Condition II events in UFSAR Chapter 15 (Accident Analysis). Since unit trips

would be less likely to occur after the installation of this modification, the modification can be considered a reliability enhancement. Since the new design is still nuclear safety related and single failure proof, the response to any accident will be unaffected and the consequences will also be unaffected. Since the effects of any single failure remain bounded by existing analyses, no malfunctions with a different result are created. The main feedwater isolation valves are containment isolation valves. The valve's seating surface, stroke times, and pressure boundary are not affected by this modification. Thus, no design bases limits for fission product barriers are affected. Also, no methods of evaluation are affected by this modification

The 10CFR50.59 analysis relating to the likelihood of occurrence of a malfunction of a system, structure, or component important to safety was addressed by a probabalistic risk analysis of failure rate data for the components involved in the modification. The conclusion was that the component failures which are introduced by this design change are not expected to make a significant contribution to the failure to isolate (estimate much less than a factor of 2).

A 10CFR50 59 evaluation concluded that this change could be made without prior approval from the NRC. No Technical Specification changes are required. No UFSAR changes are required.

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

Title: Nuclear Station Modification CN-11415/00, Addition of Vital Swing Inverters

Description: Nuclear Station Modification CN-11415/00 will add train related vital swing inverters 1EIE and 1EIF to the 120 VAC Vital Instrumentation and Control System to enhance the reliability and availability of the system. The modification will also add manual bypass switches 1EME and 1EMF for the swing inverters. New manual bypass switches 1EMAA, 1EMBB, 1EMCC, amd 1EMDD for the existing channel related inverters will be added as well. Various problems exist in the present system related to spare parts availability, concerns over inverter age related failure and the implications on unit availability associated with the possibility of loss of the 120 VAC Vital Instrumentation and Control System or its subcomponents (channels). The train related vital swing inverters will be provided with the necessary cabling, breakers, interlocks and administrative controls to preclude violating design basis. The addition of the swing inverters will allow an inoperable inverter or one ungoing preventative maintenance, to be removed from service but allow the respective panel board to return quickly to Class 1E inverter backed power (following a short transitional period on the non-safety related regulated power source (1VRD). After the implementation of this modification, the affected unit will be able to exit an LCO much more quickly upon loss of a single inverter.

Evaluation: All of the components involved in this modification are are of a quality and specification equivalent to the existing nuclear safety related equipment. The swing inverters will operate in the same manner as the existing inverters. These inverters are considered accident mitigation equipment. The inverters provide 120 VAC power to panelboards 1ERPA, 1ERPB, 1ERPC, 1ERPD which provide instrumentation and control functions. These items are shown on UFSAR Figure 8-24. Loads on these panelboards include Solid State Protection System Equipment, Nuclear Instrumentation, Auxiliary Safeguards Cabinets, Pressurizer Relief Valves, and Post Accident Recorders. All of the performance capabilities of these loads will be fulfilled by the swing inverters when they are aligned for service. Therefore the accident mitigation function of the equipment will remain unchanged. No safety analysis assumptions are affected. Since the modified design will still be nuclear safety related and single failure proof, the response to accidents will be unaffected. An analysis per 10CFR50.59 concluded that this change could be made without prior NRC approval. No Technical Specification changes are required although the Bases for Technical Specification 3.7.7 and 3.7.8 will be revised. UFSAR changes will be required for UFSAR Sections 8.3.2.1.2.1 and 8.3.2.1.2.2. UFSAR Figure 8-24 and UFSAR Table 8-11 will also be revised.

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

Title: Nuclear Station Modification CN-21404/00, "Auxiliary Building Ventilation System System Bypass Alignment"

Description: Modification CN-21404/00 will allow the Unit 2 Auxiliary Building Ventilation System filter units to operate in a filter bypass alignment during normal operation. A single failure concern was identified on this system in 1995. An interim corrective measure was to place all system flow through the filters (i e. operate in the system's accident alignment). This modification will restore control air to the Auxiliary Building Ventilation System filter inlet isolation damper, outlet isolation damper and existing bypass damper. A second filter unit bypass damper will be added in series with the existing bypass damper. This damper will have a pneumatic actuator and will receive control air from the same source as the filter unit dampers. If one bypass damper fails to close during a design basis accident, the second damper will close to ensure that air exhausted from the Auxiliary Building does not bypass the filter unit. The modification will also add a second solenoid valve in series with the existing solenoid valve for control of the filter unit dampers. Successful solenoid valve actuation of either Train A or Train B will cause all four dampers per train (two bypass dampers, one inlet damper, and one outlet damper) to change position. If a control signal is initiated to place Train A in a filtered alignment, Train B will also go to the filtered alignment and vice versa Receipt of Train A or Train B Safety Injection signal during an accident will place both trains of the Auxiliary Building Ventilation System filter units in the filtered alignment. If either Train Safety Injection signal is lost the opposite train will provide a control signal to place the filter units in the filtered alignment.

Evaluation: Since the Auxiliary Building Ventilation System is not an accident initiator, this modification cannot increase the frequency of occurrence of any accident. All components added by the modification will be of a quality consistent with the system. The realignment of the system from the bypass mode to the filtered mode occurs in a few seconds. It would be about fifteen minutes before the system would be required to mitigate the effects of recirculated ECCS flow. This modification will return the operation of the Auxiliary Building Ventilation System to its original design. A 10CFR50.59 evaluation of the modification concluded that the modification could be made without prior NRC approval. No Technical Specification changes are required. UFSAR Sections 7.6.12, 9.4.3.2, UFSAR Table 9-28 and UFSAR Figure 9-123 will be revised.

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

Title: Nuclear Station Modification CN-21413/00, De-energize the hydraulic solenoids on the Main Feedwater Isolation Valves

Description: The Main Feedwater Isolation Valves are designed to isolate feedwater flow to a faulted steam generator upon receipt of a feedwater isolation signal. The valves are also used to isolate feedwater to the Steam Generators during normal shutdown and startup operations. These valves are designated as 2CF33, 2CF42, 2CF51, and 2CF60

The two hydraulic solenoids in the operator of the Main Feedwater Isolation Valves are normally energized and are de-energized to close the valve. A single failure that de-energizes either of these solenoids will close the valve causing a trip of the associated unit. This modification is a reliability enhancement to steady state unit operation.

Modification CN-21413/00 will change the control circuitry for the hydraulic solenoid valves to a normally de-energized state. Currently, these valves are normally energized in a closed position, allowing the hydraulic oil pressure to be exposed to the bottom of the valve actuator by maintaining the vent flowpath isolated. Upon loss of power to these solenoid valves, the flowpath is aligned to the reservoir allowing the hydraulic oil pressure to decay, resulting in closure of the valve from the application of nitrogen pressure to the top of the actuator.

In addition, since the 10% stroke testing circuit associated with each train of main feedwater isolation valves has never been used, this testing circuit will be removed. This will result in removal of two control switches on Main Control Board 2MC2.

The new control circuit requires a new method to alert the operators to a loss of control power. The new circuit has two parallel, normally energized relays which contain two normally open contacts wired in series, per train that control the status of two parallel train related normally closed solenoid valves.

Features have been provided to indicate the presence of problems, such as a loss of power to a relay, which might otherwise go undetected and preclude operation of these Engineered Safety Features. The Operator Aid Computer will monitor the energized state of the relays. A de-energized relay closes its associated contact and nothing is inoperable but the plant is closer to a trip condition (solenoid valve opening - main feedwater isolation valve closure). The Regulatory Guide 1.47 Bypass Panel will monitor the control power to these valves.

Evaluation: This modification is considered an adverse change in that the design may be less fail safe, although still single failure proof, than the current design. The benefits of the de-energized solenoid design that would minimize the potential for a unit trip is balanced against additional complexities in the circuitry and the characteristic of having to energize the solenoids to perform a safety function. Since the change may be considered adverse, a 10CFR50.59 evaluation was performed.

Seven of the eight criteria of 10CFR50.59 were addressed using the following argument. A Feedwater Isolation causes a Reactor/Turbine Trip and turbine trips are evaluated as ANS Condition II events in UFSAR Chapter 15 (Accident Analysis). Since unit trips

would be less likely to occur after the installation of this modification, the modification can be considered a reliability enhancement. Since the new design is still nuclear safety related and single failure proof, the response to any accident will be unaffected and the consequences will also be unaffected. Since the effects of any single failure remain bounded by existing analyses, no malfunctions with a different result are created. The main feedwater isolation valves are containment isolation valves. The valve's seating surface, stroke times, and pressure boundary are not affected by this modification. Thus, no design bases limits for fission product barriers are affected. Also, no methods of evaluation are affected by this modification.

The 10CFR50.59 analysis relating to the likelihood of occurrence of a malfunction of a system, structure, or component important to safety was addressed by a probabalistic risk analysis of failure rate data for the components involved in the modification. The conclusion was that the component failures which are introduced by this design change are not expected to make a significant contribution to the failure to isolate (estimate much less than a factor of 2).

A 10CFR50.59 evaluation concluded that this change could be made without prior approval from the NRC. No Technical Specification changes are required. No UFSAR changes are required.

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

Title: Nuclear Station Modification CN-21415/00, Addition of Vital Swing Inverters

Description: Nuclear Station Modification CN-21415/00 will add train related vital swing inverters 2EIE and 2EIF to the 120 VAC Vital Instrumentation and Control System to enhance the reliability and availability of the system. The modification will also add manual bypass switches 2EME and 2EMF for the swing inverters New manual bypass switches 2EMAA, 2EMBB, 2EMCC, amd 2EMDD for the existing channel related inverters will be added as well. Various problems exist in the present system related to spare parts availability, concerns over inverter age related failure and the implications on unit availability associated with the possibility of loss of the 120 VAC Vital Instrumentation and Control System or its subcomponents (channels). The train related vital swing inverters will be provided with the necessary cabling, breakers, interlocks and administrative controls to preclude violating design basis. The addition of the swing inverters will allow an inoperable inverter or one ungoing preventative maintenance, to be removed from service but allow the respective panel board to return quickly to Class 1E inverter backed power (following a short transitional period on the non-safety related regulated power source (VRD). After the implementation of this modification, the affected unit will be able to exit an LCO much more quickly upon loss of a single inverter.

Evaluation: All of the components involved in this modification are are of a quality and specification equivalent to the existing nuclear safety related equipment. The swing inverters will operate in the same manner as the existing inverters. These inverters are considered accident mitigation equipment. The inverters provide 120 VAC power to panelboards 2ERPA, 2ERPB, 2ERPC, 2ERPD which provide instrumentation and control functions. These items are not shown in the UFSAR; however the equivalent Unit 1 items are shown in UFSAR Figure 8-24. Loads on these panelboards include Solid State Protection System Equipment, Nuclear Instrumentation, Auxiliary Safeguards Cabinets, Pressurizer Relief Valves, and Post Accident Recorders. All of the performance capabilities of these loads will be fulfilled by the swing inverters when they are aligned for service. Therefore the accident mitigation function of the equipment will remain unchanged. No safety analysis assumptions are affected. Since the modified design will still be nuclear safety related and single failure proof, the response to accidents will be unaffected. An analysis per 10CFR50.59 concluded that this change could be made without prior NRC approval. No Technical Specification changes are required although the Bases for Technical Specification 3.7.7 and 3.7.8 will be revised. UFSAR changes will be required for UFSAR Sections 8.3.2.1.2.1 and 8.3.2.1.2.1.4. UFSAR Figure 8-24 and UFSAR Table 8-11 will be revised. UFSAR Table 3-106 and UFSAR Table 9-35 will be revised.

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

Title: Nuclear Station Modification CN-21417/00, Rewire relay logic in the Auxiliary Feedwater Pump Turbine Control Panel to fail (de-energize) to the Remote (Main Control Room) mode of control

Description: .The Turbine Driven Auxiliary Feedwater Pump Control Panel provides the capability of controlling the Turbine Driven Auxiliary Feedwater Pump from a location outside the Control Room during a loss of Control Room event. Control is transferred from the Control Room to the Turbine Driven Auxiliary Feedwater Pump Control Panel via switches located on the Turbine Driven Auxiliary Feedwater Pump Control Panel. When control is transferred, a group of relays in the panel actuate, disabling control from the Control Room and enabling the controls on the Turbine Driven Auxiliary Feedwater Pump Control Panel.

The relays are continuously energized and any short circuit in the panel causes the associated train of Auxiliary Feedwater to fail control to the Turbine Driven Auxiliary Feedwater Pump Control Panel. Within the past several years there have been three unplanned transfers to the Turbine Driven Auxiliary Feedwater Pump Control Panel as a result of relay failures within the Turbine Driven Auxiliary Feedwater Pump Control Panel. If such a failure were to occur during a design basis event, it would hamper the ability of Operations to effectively control the plant and mitigate an accident. During a Steam Generator Tube Rupture Event, such a failure leads to overfill of the Steam Generator and subsequent release of reactor coolant system fission products through the steam generator PORVs and/or secondary side code safety relief valves.

This modification will achieve the following:

1. Protect the transfer switches with a clear plastic cover to prevent inadvertent transfer to the panel (the switch has been bumped in the past during maintenance and control transferred).

2. Reconfigure the Turbine Driven Auxiliary Feedwater Pump Control Panel such that the panel is de-energized during normal operation. This will prevent spurious transfers due to electrical faults within the Turbine Driven Auxiliary Feedwater Pump Control Panel.

Evaluation: There does not appear to be any reason that the Turbine Driven Auxiliary Feedwater Pump Control Panel cannot be reconfigured to allow a de-energized design. The reason for the current energized design may have been to provide a fail safe fire protection design prior to committing to the dedicated Safe Shutdown System for fire response. Since the Safe Shutdown System is the committed post fire safe shutdown facility for the worst case fire at Catawba, the Auxiliary Shutdown Complex is not needed for passive fire response. It can be used for certain fires, if available, and is preferable since it has more controls and indication than the Safe Shutdown System which includes the Safe Shutdown Facility.

Reconfiguring the Turbine Driven Auxiliary Feedwater Pump Control Panel to a de-energized state will have reliability benefits by eliminating spurious transfers due to certain electrical faults within the panel.

The systems, structures and components affected by this modification are nuclear safety related. All rewiring will be within the Turbine Driven Auxiliary Feedwater Pump Control Panel. No field wiring changes will be required. A Safe Shutdown Modification Screening Review has been performed. The Safe Shutdown Design Basis is hot standby per UFSAR 5.4.7.2.6. The ability of the plant to achieve Hot Standby and proceed to Cold Shutdown is not adversely affected by the modification. The revised scheme does not introduce any electrical control anomalies.

The response of the Turbine Driven Auxiliary Feedwater Pump to UFSAR accidents for which it is designed to mitigate, is unaffected by this modification. Control Room controls of the Turbine Driven Auxiliary Feedwater Pump and the starting signals that are related to UFSAR accident mitigation (Blackout and Lo-Lo Steam Generator levels on 2 of 4 Steam Generators are not affected by this modification. There is an interlock whih prevents a Turbine Driven Auxiliary Feedwater Pump start upon the prescence of a safety injection signal simultaneous with a blackout. All design criteria continue to be met after this modification.

No Technical Specification changes are required. No UFSAR changes are required. A 10CFR50.59 evalaution concluded that this change could be made without prior NRC approval.

7 Type: Procedure **Unit:** 1

Title: Procedure OP/1/A/6100/003 Revision 88 " Controlling Procedure for Unit Operation".

Description: Operations Procedure OP/1/A/6100/003 Revision 88 "Controllıng Procedure for Unit Operation" is being revised to allow an alternate method of operation of Unit 1 near the end of cycle 13. This will result in a Tavg coastdown whereby the reactor will be "off program". The procedure change adds a new Enclosure 4.4 and revises Enclosures 4 2 and 4.4 to explicitly define operating limits and termination crıteria to assure compliance with safety analysis assumptions The purpose of this operation method is to get more power out of the reactor by running at colder temperatures relative to the normal temperature/power relationship shown in UFSAR Figure 4-76 "Unit 1 Reactor Coolant System Temperature - Percent Power Map". The loss of excess reactivity due to fuel burn-up is compensated for by the positive reactivity introduced by the colder reactor coolant temperatures combined with the negative moderator temperature coefficient (MTC). This method of operation allows for extending the end of cycle burn-up date (cycle end date) and accommodates business flexibility needs. Other evaluations have provided a basis upon which all pertinent issues have been evaluated and limıts placed procedurally, where necessary, to assure the operation of Catawba Unıt 1 remains within its licensing basis.

The Tavg coastdown is described ın three phases. Phase I involves opening the fourth Turbine Control Valve in an attempt to keep power near 100% as temperature drops (~577 degrees F.) The end of Phase I has the Reactor and Turbine near 100% at a colder Tavg. Phase II maıntains the fourth Turbine Control Valve constant and allows Tavg to drop with a corresponding drop in steam flow and Turbine/Reactor power (~94% Rated Thermal Power, 570 degrees F.) Phase III involves closing the Turbine Control Valve to reduce power and could be considered a "normal progression to shutdown except for the "off program" conditions from the prior Tavg drop. This period will take the Unit to a power level not to be less than 70% power. Phase III will not require any further adjustments of Tref as this will occur (decrease) as the Turbine Power is reduced via Control Valve closure. The "off program" terminology describes a deviation from UFSAR Figure 4-76 Temperature/Power relationship and not the need for adjustment of Tref to avoid rod motion or alarms. It is the initial conditions of the system that are important and that is determined by quantifying the variance with respect to UFSAR Figure 4-76

This evaluation is for the Operations Procedure which allows for operation of Catawba Unit 1 in a manner not previously experienced. Prior end of cycle operation has included a "power coastdown" as described in UFSAR Section 4.3.1.1 "Fuel Burnup" which involves reducing power along the normal operating space as defined in UFSAR Figure 4-76 The difference in this "Tavg coastdown" method is the Unit will actually be operated at colder "off program" temperatures than previously defined by the relationship of UFSAR Figure 4-76. The implications of reduced temperature operation have been considered. Appropriate technical evaluations have been performed to assure that all structures, systems and components which may experience different conditions than those which they have been routinely subjected, can perform their intended design functions within the safety envelope to which the plant was originally licensed.

Procedural limits have been placed in Enclosures 4.2, 4.3, and 4 4 of procedure

OP/1/A/6100/003 revision 88 to assure the Unit is operated within analyzed initial conditions from which any design basis challenges can initiate. A new Enclosure 4.4 and revisions to existing enclosures have been added to explicitly define operating limits to assure compliance with safety analysis assumptions.

Evaluation: A 10CRF50.59 evaluation determined that this procedure change could be made without prior NRC approval. No Technical Specification changes are required. Changes will be required for UFSAR Chapters 4 and 15 to describe this alternate method of "off program" operation.

22 **Type:** Procedure **Unit:** 0

Title: Procedure OP/1/A/6450/017 Revision 52A, Enclosure 4.9, "Abnormal Air Release Mode"

Description: During normal plant operation, corrective maintenance activities or unexpected equipment or component failures may result in the inability to reduce containment pressure through the normal air release path. For these abnormal situations, an alternate air release method is necessary to remain within the Technical Specification 3.6.4 containment pressure limits of -0.1 and 0.3 psig. This evaluation will determine if releasing air through the air addition path is acceptable. This alternate air release method involves placing a jumper within the electrical control circuitry to maintain valve 1VQ-13 open, placing the 1VQ-13 control switch in AUTO, and then manually opening the containment isolation valves, 1VQ-15B and 1VQ-16A from the control room. Air would then be forced out of the upper containment and into the Elevation 543 Mechanical Penetration Room by the pressure differential. The airflow from the penetration room would then be exhausted through the Auxiliary Building Ventilation System filtered exhaust trains to the unit vent. Monitoring the release would be performed using the containment radiation monitor, EMF-39(L), or the unit vent radiation monitor, EMF-36(L). Auxiliary Building radiation levels would be monitored by EMF-41 and/or manually as required by Radiation Protection. The maximum release airflow rate conservatively calculated using a 0.3 psid would be less than 200 cfm As the containment pressure decreases, the airflow rate would also decrease The Gaseous Waste Release volume will be conservatively calculated using a flow rate of 350 cfm, which is the same methodology applied when the totalizer in the normal air release path is inoperable.

Evaluation: Technical Specification 3.6.3 "Containment Isolation Valves", allows opening of the Containment Air Release and Addition System containment isolation valves for pressure control, ALARA, air quality considerations, or surveillances. This procedure change alters the way a containment air release will be performed to reduce containment pressure as described in UFSAR Section 9.5.10. This alternate air release or pressure control method will only be used in abnormal or emergency situations. The alternate containment pressure release through the air addition path will be accomplished by releasing air from the upper containment, similar to normal containment pressure reductions. Since this alternate air release method would only be used in rare situations, it will not be described in the UFSAR. The ability of the Containment Air Release and Addition System containment isolation valves to perform their safety related design basis function will not be affected by this alternate air release method. The air released through this alternate path will be filtered by the Auxiliary Building Ventilation System prior to release through the unit vent and the design basis function of the Auxiliary Building Ventilation System will not be affected. Sufficient controls are established within the procedure to ensure containment pressure remains within the limits of Technical Specification 3.6.4 "Containment Pressure". Radiological sampling and monitoring of releases will be the same as normal releases. The methodology used to calculate any dose rates or total doses will not be changed by this procedure. Therefore, there is no effect on the 10CFR20, 10CFR50 Appendix I, and 40CFR190 licensing bases. Tornado protection measures will be procedurally controlled. No Technical Specification changes are required. No UFSAR changes are required. A 10CFR50.59 evaluation concluded that this change could be made without prior NRC approval.

27 **Type:** Procedure **Unit:** 0

Title: Procedure PT/0/A/4150/012 B, "Moderator Temperature Coefficient of Reactivity Measurement (EOL)", Revision 14

Description: Revision 14 to Procedure PT/0/A/4150/012 B, "Moderator Temperature Coefficient of Reactivity Measurement (EOL)", adds the ability to use the steam dump to condenser system to create an "artificial" steam load on the reactor to maintain the reactor near full power during test. The use of steam dumps in this manner is not explicitly described in UFSAR; the steam dump system is described as providing an artificial load to allow the reactor to respond to a sudden loss of load without a reactor trip.
Performance of measurement near full power minimizes the corrections required to compare measured moderator temperature coefficient (MTC) to the reference conditions (hot full-power, 300 ppm boron, equilibrium xenon and samarium). Without use of steam dumps to provide additional steam load, reactor power must be reduced to ensure sufficient capacity exists on the turbine control valves to accommodate a reactor coolant system cooldown of approximately 5 degrees. Eliminating the power decrease increases measurement accuracy and reduces difficulties with planning and scheduling of the measurement.
General description of steps to perform EOL MTC measurement:
1) Measure/record full power, steady state reactivity parameters (boron concentration, temperature, burnup, xenon worth, control rod position)
2) Borate reactor coolant system to achieve 5 degree cooldown
3) Open turbine control valves as steam pressure decreases, maintain full power steam flow
4) If turbine control valves reach full open, use steam dumps to condenser for additional steam flow to maintain full power steam flow.
5) Establish steady state conditions at reduced temperature
6) Measure/record reactivity parameters
7) Dilute reactor coolant system to return to normal operating temperature
8) Close steam dump valves to maintain full power steam flow
9) When steam dumps are closed, close turbine control valves to maintain full power steam flow
10) Establish steady state conditions at normal temperature
11) Measure/record reactivity parameters
12) Determine MTC from reactivity data collected at the three statepoints and apply corrections for comparison to surveillance limit.

Evaluation: The turbine bypass system and steam dump control systems are described in UFSAR Sections 10.4.4 and 7.7, respectively. The purpose of the systems is to:
1) create an artificial load on reactor following large turbine load reductions,
2) remove decay heat following a reactor trip, and
3) maintain the unit in hot standby condition.
The steam dump system is not essential for the safe shutdown of the unit, and is not designated as safety related. Failure of the system does not preclude operation of any system essential for safe shutdown.
The relevant accident categories are "Increase in Heat Removal by the Secondary System" (UFSAR Section 15.1) and "Decrease in Heat Removal System by the Secondary System" (UFSAR Section 15.2).
UFSAR Section 15.1.3 describes excessive increase in secondary steam flow. An

excessive increase is defined as a rapid increase in steam flow that causes a power mismatch between reactor and steam generator load UFSAR Section 15.1.4 is the inadvertant opening of steam generator relief or safety valve. Inadvertant opening of a steam dump valve is included in this section
The steam dump system is non-safety, but is important to safety. The capacity of steam dump was considered when selecting the power level for turbine trip causes reactor trip permissive (P-9).
The use of steam dumps to provide additional steam flow capacity to perform moderator temperature coefficient measurement results in a minimal increase in the frequency of the increase in heat removal accident and no increase in the frequency of the decrease in heat removal accident. There is no increase in the likelihood of the occurrence of a malfunction of an SSC important to safety, and no effect on the consequences of an accident or a malfunction of an SSC. UFSAR analyses (sections 15.1 and 15 2) remain bounding, no new malfunctions or accidents beyond those covered in sections 15.1 and 15.2 are created Design basis limits and methods of evaluation described in the UFSAR are unaffected.

No Technical Specification changes are required No UFSAR changes are required A 10CFR50 59 evaluation concluded that this procedure change could be made without prior NRC approval.

29 **Type:** Procedure **Unit:** 1

Title: Procedure PT/1/A/4200/001N Revision 44, Reactor Coolant System Pressure Boundary Valve Leak Rate Test

Description: The purpose of procedure PT/1/A/4200/001N is to verify that the leakage past any Reactor Coolant System Pressure Boundary Valve does not exceed the value specified in Technical Specification 3.4.14, by satisfying the requirements of Technical Specification Surveillance Requirement 3.4.14.1. The valves that are tested by this procedure are important in preventing overpressurization and rupture of the Emergency Core Cooling System 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 lines to the Reactor Coolant System cold legs and hot legs, as well as the Residual Heat Removal System suction isolation valves in the "B" and "C" hot legs.

This procedure change is to specify valves within the test boundary which must be returned to the "as found" position prior to entering Mode 2.

In a previous procedure revision, the "Valve Checklist" for system alignment during the Pressure Boundary Valve Test was separated into two enclosures. One enclosure entitled "Valve Checklist - ECCS Operating", included valves operated from the Control Room and Safety Injection System Test Panel. System re-alignment per this valve checlklist must be re-established prior to entering Mode 3 to ensure ECCS operability per Technical Specification 3.5.2. Another enclosure entitled "Valve Checklist - Safety Injection Test Header", identifies manual valves located in the Auxiliary and Reactor Buildings which have no influence on ECCS operability. A caution statement in the "Valve Checklist - Safety Injection Test Header" enclosure presently states that valves must be returned to their "as found" position prior to entering Mode 3. Since valves identified on this procedure enclosure do not affect operability of ECCS functions as described in Technical Specification 3.5.2, the caution statement will be revised to state that valves shall be returned to the as-found position prior to entering Mode 2.

Evaluation: This procedure change provides a scheduling enhancement which has no negative effect on the system alignment for the test. There is no effect on any accident analyzed in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required. A 10CFR50.59 evaluation concluded that this change could be made without prior NRC approval.

14 **Type:** Procedure **Unit:** 1

Title: Procedure PT/1/A/4450/001D Revision 26

Description: Procedure PT/1/A/4450/001D is being revised to allow the Containment Purge Ventilation System filter units to be tested while in Modes 1,2,3 or 4. Containment Purge Ventilation System filter unit testing during Modes 1,2,3 or 4 will reduce outage testing duration and minimize potential delays for maintenance during the Outage.
During the proposed testing, the Containment Purge Ventilation System filtered exhaust duct inlet access doors (located downstream of the Containment Isolation Valves and Reactor Building isolation damper (1CPE D-1) will be opened. Air from the 594' elevation of the Auxiliary Building will be pulled into the filtered exhaust ducting. The air will then pass through the Containment Purge Ventilation System filter units and be exhausted through the filtered exhaust ducting outlet access doors. The outlet access doors are located downstream of the Containment Purge Ventilation System filter units and prior to the filtered exhaust backdraft damper (lCPE D-6). Access doors for both the inlet and outlet ductwork were sized using the flow orifice loss equation Q=AoCd(2gh)^ 1/2. The access door openings were sized to match calculated ductwork static pressure losses in calculation CNC-1211.00-00-0039. The duct access doors will be opened and closed per the work order process using approved maintenance procedures. Airflow will be throttled using a sliding access door. The filtered exhaust backdraft damper (lCPE D-6) will be tied in the closed position by maintenance procedure MP/0/A/7450/048 (Temporary Alterations of Station Dampers) to recirculate air back into the Auxiliary Building and isolate the exhaust air from the Unit Vent. Only the Containment Purge Ventilation System filtered exhaust fans (CPXF-1A and CPXF-1B) will be operated during Modes 1,2,3 or 4. Electrical jumpers will be placed within control panel 1 RB ECP-l to allow these fans to operate. Sliding links within this electrical panel will be opened to prevent remaining portions of Containment Purge Ventilation System from operating including the duct heater (lHETR0528). Operation of the duct heater will be controlled by installation of a jumper. The duct heater will only be placed in operation for implementation of Enclosure 13.13. The sliding links will prevent the Containment Purge Supply Fans from starting, Containment Isolation Valves from opening, and Reactor Building Isolation Dampers (lCPS D-7, lCPE D-l) from opening. In addition the motor control center breakers will be opened for Containment Purge supply fans (CPSF-1A, CPSF-1B) which will also prevent the above components from operating. With these procedural controls implemented, the containment isolation valves will remain sealed closed in accordance with Technical Specification 3.6.3 while the exhaust fans are placed in operation to support testing in Modes 1,2,3 or 4.

Other minor changes to the procedure include the addition of Procedure OP/0/A/6700/010 as an aid to operate the Air Data Multimeter (ADM), as well as deletion of U-tube manometers, inclined manometers, and psychrometers as test equipment. ADMs are now used to perform these testing functions. PT/1/A/4450/001D was also administratively revised to the ensure that the Incore Instrument Filter Unit (IIFU-1) continues to be tested only during Outage Modes 5 and No Mode. Replacement clamp on ammeters have been added to perform the heater dissipation test. Previous ammeters are no longer available. Replacement ammeters are equivalent to the replacement models. The clamp on ammeters have been evaluated as acceptable per an engineering. Per the manufacturer's information, the accuracy of the ACDC 1001 is equivalent to that of the ACDC 1000. All testing equipment including the new DOP

generator (ATI Model TDA-5B) meets the requirements of ANSI N510-1980. The Containment Purge Ventilation System exhaust will only be temporarily configured to support testing in Modes 1,2,3 and 4. The procedure contains adequate guidance to return the system to its normal configuration. Therefore, system operation and design functions as described in the SAR will not be affected.

Evaluation: Testing of the Containment Purge Ventilation System filter units in Modes 1,2,3 and 4 will be completed to allow air in the Auxiliary Building to be pulled through the filter units by the exhaust fans and then exhausted back into the Auxiliary Building. Electrical links in control panel 1 RB ECP-1 will be opened to maintain Containment Isolation Valves (CIVs) in the closed position. Control Room 1MCB 5 selector switches and push buttons for CIVs will be maintained in the "Blocked Closed" position. Control power will be removed from MCC breakers associated with the Reactor Building isolation dampers to ensure that these dampers remain in the closed position. Steps have been added to the testing procedure to ensure that the test is stopped if the air temperature entering the filtered exhaust reaches or exceeds 110 degrees F. This will ensure that the high temperature limits of the Auxiliary Building, as identified in the Environmental Qualification Criteria Manual (EQCM), are not exceeded. The filtered exhaust backdraft damper (1CPE D-6) will be tied in the closed position during testing to isolate exhaust airflow from the Unit Vent.

The Containment Purge Ventilation System is not an ESF system and thus NRC Regulatory Guide 1.52 is not applicable to the system However, since some credit is taken for the system in the Fuel Handling Acccident dose analysis, a comparison is made to NRC Regulatory Guide 1.52 in UFSAR Table 12-28 to provide reasonable assurance that the system is designed and maintained in a quality manner.

In conclusion, these procedure changes will support filter testing activities in Modes 1,2, 3 and 4 and will not adversely affect the Containment Purge Ventilation System as described in the UFSAR. After testing in Modes 1,2,3 and 4 is completed, the Containment Purge Ventilation System will be returned to its normal design configuration. The proposed tests will not affect any analyses as described in the UFSAR. Testing of filters in Modes 1,2,3 and 4 will ensure that the Containment Purge Ventilation System continues to meet system design basis requirements identified in Technical Specifications 3.9.3 and 5.5.11.

No Technical Specification changes are required A 10CFR 50.59 evaluation concluded that this procedure change could be made without prior NRC approval. No UFSAR changes are required.

19 **Type:** UFSAR Change **Unit:** 0

Title: Change to UFSAR Section 10.3 2 and 10.4.9.2 to address Auxiliary Feedwater Pump Turbine Steam Supply Piping Heat Trace and Temperature Monitoring Instrumentation

Description: UFSAR Sections 10.3.2 and 10.4.9.2 are being changed to address the Auxiliary Feedwater Pump Turbine Steam Supply Piping Heat Trace and Temperature Monitoring Instrumentation. A new Selected Licensee Commitment (SLC) "Auxiliary Feedwater Pump Turbine Steam Supply Piping Temperature Monitoring Instrumentation and Heat Trace System" is being implemented Currently the heat trace system has not been adequately evaluated for inclusion in the UFSAR. This evaluation will examine the operation of the Auxiliary Feedwater Pump Turbine steam supply piping heat trace and associated pipe temperature monitoring instrumentation, in conjunction with the administrative controls provided by the new SLC and existing plant procedures and programs.

The Auxiliary Feedwater Pump Turbine is the driver for the Turbine- Driven Auxiliary Feedwater Pump. The Auxiliary Feedwater Pump Turbine steam supply piping is part of the Main Steam to Auxiliary Equipment System. The Auxiliary Feedwater Pump Turbine steam supply piping heat trace is part of the Electrical Heat Tracing System. This evaluation demonstrates how operability and reliability of the Turbine Driven Auxiliary Feedwater Pump are ensured by the function of the Main Steam to Auxiliary Equipment System heat trace equipment when operated within the administrative requirements set forth in the new SLC. Both Catawba units are equipped with essentially identical systems.

The Turbine Driven Auxiliary Feedwater Pump is part of the Auxiliary Feedwater System, which is designed to remove reactor decay heat. The system is designed to allow a cool down of the plant from normal operating temperatures and pressures to the point that Residual Heat Removal may be placed in service, thus assuring long term decay heat removal. The Turbine Driven Auxiliary Feedwater Pump also functions as part of the Standby Shutdown System. In this role, the Turbine Driven Auxiliary Feedwater Pump is required to remove decay heat while maintaining stable operating conditions close to normal operating temperatures and pressures until sufficient repairs are made to allow a normal cool down from the Control Room or the Auxiliary Shutdown Panels. The turbine receives steam from electrically heat traced Main Steam to Auxiliary Equipment System piping originating from the B and C Steam Generators. The Turbine Driven Auxiliary Feedwater Pump is a nuclear safety related component, which receives Engineered Safeguards System auto-start signals on loss of offsite power, loss of all emergency AC power, and steam generator low-low level on 2 out of 4 steam generators. The Turbine Driven Auxiliary Feedwater Pump is required to mitigate several UFSAR Chapter 15 accident scenarios and is designated as "Very Important" with respect to core damage mitigation per the Catawba Probabilistic Risk Assessment (PRA). The Turbine Driven Auxiliary Feedwater Pump is required to be operable per Catawba Technical Specification 3.7.5. The Turbine Driven Auxiliary Feedwater Pump and the Auxiliary Feedwater System are described in Sections 1.8, 7.4.1, and 10.4.9 of the UFSAR.

The Auxiliary Feedwater Pump Turbine steam supply piping is made up of ASME Class II and Class III, six inch diameter schedule 80 piping. The piping is predominantly carbon steel. However, a short run of stainless steel pipe has been installed on Unit 1. The Auxiliary Feedwater Pump Turbine steam supply piping ties into the B and C Steam

Generator main steam lines in the Inside Doghouse. The Auxiliary Feedwater Pump Turbine steam admission valves are also located in the Inside Doghouse. From the doghouse, the piping runs down into the Turbine Driven Auxiliary Feedwater Pump pit. The piping is nuclear safety related. The design temperature of the piping is 600 degrees F. The Main Steam to Auxiliary Equipment System piping has limited condensate drainage capacity provided by two drain orifices.

The Main Steam to Auxiliary Equipment System piping is equipped with an electrical heat trace system and insulation to minimize the volume of steam condensation generated during Auxiliary Feedwater Pump Turbine start-up operation. Large amounts of condensate, generated in the process of heating up the Main Steam to Auxiliary Equipment System steam lines, could result in such problems as turbine over-speed trips, turbine damage or water hammer. On both Unit 1 and Unit 2, the Main Steam to Auxiliary Equipment System heat tracing is divided into ten individually controlled circuits located between the steam admission valves, SA-2 and SA-5, and the turbine. A circuit consists of two heat trace cables designated as "primary" and "backup". The two cables operate concurrently, regardless of the pipe temperature. Each heat trace controller is equipped with an LED temperature display that can be used for local monitoring of the pipe temperatures. The controlling setpoint for each heat trace circuit is 500 degrees F with an allowable range of 495 degrees F to 505 degrees F. The entire Main Steam to Auxiliary Equipment System heat trace system is made up of non-safety components. The Main Steam to Auxiliary Equipment System heat trace system receives power from a 600V Motor Control Center via a 120V AC Trace Heating Panelboard, which is a non-safety grade source. The heat trace is required to be functional in Modes 1, 2, and 3 when the Auxiliary Feedwater Pump Turbine is in the standby-readiness mode. Engineering calculations have concluded that sufficient orifice drainage capacity exists to handle condensate generated during Auxiliary Feedwater Pump Turbine steady-state operations without functional heat trace.

Each section of Main Steam to Auxiliary Equipment System piping served by a heat trace circuit is equipped with a thermocouple, which provides temperature indication on a chart recorder located in the Auxiliary Building. The chart recorder thermocouples are independent from the thermocouples in the heat trace control circuits. The chart recorder provides a local display of ten channels of Main Steam to Auxiliary Equipment System pipe temperature, as well as local indication of low and high pipe temperature alarm conditions. In addition, the chart recorder provides a generic alarm signal to an annunciator located in the control room. The low and high temperature alarm setpoints for the chart recorder are 385 degrees F and 560 degrees F. respectively. The chart recorders are not nuclear safety related components. The chart recorder receives power from a source that is not nuclear safety related. Operations checks the chart recorder for alarm status and general functionality on a once per shift basis.

The generic alarm signal generated by the chart recorder is fed to a reflash module. The reflash module consolidates several potential heat trace related alarm conditions into one outgoing alarm signal to an annunciator window located in the control room An alarm signal generated by the Main Steam to Auxiliary Equipment System pipe temperature chart recorder would result in the illumination of an indicator light located on the exterior of the panel. The indicator light is labeled such that source of the alarm can be identified. The Reflash Cabinet is made up of components that are not nuclear safety related. The cabinet receives power from battery backed 125 VDC Distribution Center CDA which is

a reliable (but not nuclear safety related) source.

An outgoing alarm signal from the Reflash Module results in the illumination of an annunciator window in the Control Room. The lamp box and associated Annunciator Cabinet are components that are not nuclear safety related. They receive power from battery backed 125 VDC Distribution Center CDB, a reliable (but not nuclear safety related) source.

The 500 degree F high temperature alarm setpoint was established to protect against exceeding the design temperature (600 degrees F) of the Main Steam to Auxiliary Equipment System piping. Exceeding the piping design temperature (depending on the final temperature and duration) could potentially damage the piping and/or supports and ultimately result in the inoperability of the Turbine Driven Auxiliary Feedwater Pump. The most likely scenario for reaching or exceeding the high temperature setpoint is a section of heat trace failing to de-energize on high temperature. For this scenario, the piping design temperature would eventually be exceeded. Upon receipt of the annunciator in the Control Room, due to a high temperature alarm, operators are required, per the annunciator response procedure, to open the feeder breaker for the section of heat trace associated with the high temperature section of pipe An engineering calculation determined that the maximum allowed operator response time for opening a Main Steam to Auxiliary Equipment System heat trace feeder breaker as a result of a high Main Steam to Auxiliary Equipment System pipe temperature alarm is 40 minutes. The 40 minute time constraint is in place to ensure that the pipe design temperature is not reached. For conservatism, the time limit is reflected in the annunciator response procedure as 30 minutes.

The 385 degree F low temperature alarm setpoint was established to support the Main Steam to Auxiliary Equipment System pipe low temperature limit of 375 degrees F. The 375 degree F limit was established to minimize the volume of condensate generated during Auxiliary Feedwater Pump Turbine start-up and thus maintain operability of the Turbine Driven Auxiliary Feedwater Pump. Large volumes of condensate in the Main Steam to Auxiliary Equipment System piping could result in turbine over-speed trips, water hammer or turbine damage. All ten Main Steam to Auxiliary Equipment System piping sections per unit are required to have an indicated temperature of 375 degrees F or greater for the Auxiliary Feedwater Pump Turbine to be considered operable without further evaluation. In the event the indicated temperature of one or more sections of the piping drops below 375 degrees F, Operations performs the "Heat Trace Verification" enclosure in the Auxiliary Feedwater System operating procedure. A procedure enclosure instructs the operators to obtain manual temperature readings on the surface of the low temperature pipe to confirm the low temperature condition. Despite the temperature indication, it is possible to consider the Auxiliary Feedwater Pump Turbine operable if actual pipe temperatures are found to be at 375 degrees F. or above.

A new Selected Licensee Commitment entitled "Auxiliary Feedwater Pump Turbine Steam Supply Piping Monitoring Instrumentation" has been developed to ensure the operability of the Turbine Driven Auxiliary Feedwater Pump by mandating the requirements for operating and maintaining the Auxiliary Feedwater Pump Turbine steam supply piping temperature monitoring and heat trace systems. Per the requirements of the SLC, Catawba will commit that the Auxiliary Feedwater Pump Turbine steam supply piping temperature monitoring instrumentation (chart recorder and thermocouples) and

associated control room annunciator (reflash module and lamp box) will be operable. In addition, the heat trace controller temperature display function will be operable. The commitment will be applicable in the same modes that Turbine Driven Auxiliary Feedwater Pump is required to be operable (Modes 1, 2, and 3). The commitment also specifies that the chart recorder high temperature alarm setpoint shall be set to ensure that the design temperature of the steam supply piping (600 degrees F) is not exceeded. The SLC specifies that the chart recorder low temperature alarm setpoint shall be set such that a low temperature condition is identified prior to the pipe reaching the minimum temperature required for Turbine Driven Auxiliary Feedwater Pump operability (375 degrees F).

The new SLC has six remedial actions. First, if the control room annunciator for Auxiliary Feedwater Pump Turbine
steam supply piping temperature is inoperable, actions must be immediately initiated by station personnel at the local chart recorder to monitor pipe temperature data at a minimum frequency of once per 30 minutes. The 30 minute frequency supports the 40 minute maximum allowed operator response time for a high temperature piping alarm. Second, if the temperature display function of the chart recorder is inoperable, actions must be immediately taken to restore the chart recorder to operable status. Third, if a heat trace controller temperature display function is inoperable, actions must be immediately initiated to return the temperature display function to operable status. Fourth, if both the control room annunciator, and local chart recorder temperature display functions for Auxiliary Feedwater Pump Turbine steam supply piping temperature are inoperable, the chart recorder must be restored to operable status within 72 hours and actions must be immediately taken by station personnel at the heat trace controller temperature display to monitor pipe temperatures at a minimum frequency of once per 30 minutes. Fifth, if the annunciator, chart recorder and heat trace controller temperature display functions are all inoperable, the Turbine Driven Auxiliary Feedwater Pump shall be immediately declared inoperable in accordance with Technical Specification 3.7.5. The Auxiliary Feedwater Pump Turbine steam supply must be isolated and the steam supply piping heat trace de-energized - both within 40 minutes. Finally, if the chart recorder alarm setpoints are not set to support the design temperatures defined in the commitment, then actions must be immediately taken to restore the setpoints and personnel must be stationed at the chart recorder to monitor the pipe temperature display. The remedial actions ensure that the Turbine Driven Auxiliary Feedwater Pump will be taken out of service if the integrity of the Auxiliary Feedwater Pump Turbine steam supply piping, with respect to the pipe temperatures, can not be verified.

The new SLC specifies four testing requirements. First a visual check of the chart recorders for general operability and the presence of any previously unidentified alarm conditions will be accomplished once per 12 hours. Second, the chart recorders, including the control room annunciator, will be calibrated/tested at a frequency of once per 18 months. This instrumentation calibration will serve to ensure that the chart recorder and thermocouples accurately and reliably monitor and record the piping temperatures. Also, the heat trace controllers will be calibrated at a frequency of 18 months. Third, the SLC requires that the heat trace circuitry be inspected visually and via thermography at a frequency of 18 months. These requirements will serve to ensure that the heat trace system operates reliably within the specified temperature band. Potential problems in the heat trace control system, that could lead to low or high pipe temperatures would be identified and corrected. Fourth, the Auxiliary Feedwater Pump Turbine steam supply

piping shall be inspected for damage due to high temperatures once per refueling outage. In the event that the steam supply piping is unknowingly damaged during an innage period, this inspection will serve to identify and correct any problem areas.

Evaluation: A 10CFR50.59 evaluation concluded that a license amendment is not required for operation of the Auxiliary Feedwater Pump Turbine steam supply heat trace system, in conjunction with the SLC entitled " AFW Pump Turbine Steam Supply Piping Temperature Monitoring Instrumentation and Heat Trace System", and other plant administrative requirements No technical specification changes are required. A description of the Auxiliary Feedwater Pump Turbine steam supply heat trace system, and related administrative requirements, will be included in UFSAR Sections 10.3.3 and 10 4.9.2.

Neither the Auxiliary Feedwater Pump Turbine nor the Turbine Driven Auxiliary Feedwater Pump are accident initiators, as described in Chapter 15 of the UFSAR. Per Chapter 15 of the UFSAR, the Auxiliary Feedwater System is required to respond to several design basis events, including: Main Feedwater Line Rupture, Main Steamline Rupture, Small and Large Break LOCAs, Control Rod Ejection, Loss of Normal Feedwater/LOOP, Steam Generator Tube Rupture, Locked Reactor Coolant Pump Rotor and Uncontrolled Single Rod Withdrawal. However, Main Steamline Rupture is the only UFSAR accident relevant to this question with respect to evaluation of the Auxiliary Feedwater Pump Turbine steam supply piping heat trace.

Section 15.1.5 of the UFSAR discusses steam system piping failures. Catawba is analyzed for a rupture of a main steam line immediately downstream of the steam generator exit flow restrictor (upstream of the main steam isolation valves). The location of the pipe rupture would be in containment or in one of the doghouses. The six inch diameter Auxiliary Feedwater Pump Turbine steam supply piping ties into the 34 inch B and C Steam Generator main steam lines, upstream of the respective main steam isolation valve in the Inside Doghouse. Conceivably, a malfunction of the Main Steam to Auxiliary Equipment System heat trace coincident with a malfunction of the Main Steam to Auxiliary Equipment System pipe temperature monitoring instrumentation could result in a rupture of the Auxiliary Feedwater Pump Turbine steam supply piping. Per UFSAR Section 15.1.5.1, rupture of the Auxiliary Feedwater Pump Turbine steam supply piping would be considered a "minor secondary system pipe break", and is bounded by the main steamline break analysis.

If a section of heat trace were to fail in the energized state, coincident with a failure of the high temperature alarm function, the design temperature of the steam supply piping could be exceeded by several hundred degrees. A subsequent Turbine Driven Auxiliary Feedwater Pump start could then possibly result in a rupture of the piping. However, this sequence of events is unlikely. First the testing and maintenance requirements set forth in the new SLC will serve to minimize the potential for a heat trace malfunction. The current heat trace system, installed in the 1996 - 1997 timeframe, has never exhibited a failure of this nature. Second, the new SLC commitment requiring the operability and testing of the Auxiliary Feedwater Pump Turbine steam supply piping temperature monitoring instrumentation and associated annunciator will serve to minimize the potential for malfunction of this equipment. The chart recorder has never exhibited a failure in which pipe temperatures were not displayed, or alarms were not generated when required. Third, per the Auxiliary Feedwater Operating Procedure, Operations is required to check the

Main Steam to Auxiliary Equipment System pipe temperatures at the chart recorder prior to every Turbine Driven Auxiliary Feedwater Pump start. Thus, Operations would have the opportunity to identify any high piping temperature, or chart recorder malfunction, prior to starting the pump. Only an auto-start of the Turbine Driven Auxiliary Feedwater Pump could occur without the pipe temperatures having been checked immediately prior to the start. An auto-start of the Turbine Driven Auxiliary Feedwater Pump is an infrequent event - occurring only upon a station blackout or on a 2 of 4 channel low-low level on 2 of 4 steam generators. In addition, the visual check for chart recorder operability, required by the SLC further decreases the chances of a high pipe temperature condition going undetected. Based on this evidence, there is not more than a minimal increase in the frequency of a steam line break due to the scenario where a section of Main Steam to Auxiliary Equipment System heat trace fails in the energized state prior to a Turbine Driven Auxiliary Feedwater Pump start.

For the case where a heat trace section fails in the energized state after a Turbine Driven Auxiliary Feedwater Pump start, there are several possibilities. First, in response to a control room annunciator for high Auxiliary Feedwater Pump Turbine steam supply piping temperature, it would be acceptable for Operations to open the feeder breaker for the failed heat trace. As stated previously, the heat trace is not required to be functional while the Turbine Driven Auxiliary Feedwater Pump is running. Also, even if the failed heat trace remained energized, it is unlikely that the pipe temperature would ever reach the pipe design temperature while the pump was running. Per UFSAR Figure 15-30, the maximum steam line pressure case for a turbine trip is approximately 1280 psig. The corresponding temperature for saturated steam is approximately 575 deg F. Thus, the steam, being cooler than the pipe design temperature, would remove the heat added by the heat trace such that the design temperature would not be exceeded.

If a section (or sections) of heat trace were to fail in the de-energized state coincident with a failure of the low temperature alarm function, large volumes of condensate could be generated in the Auxiliary Feedwater Pump Turbine steam supply piping on a Turbine Driven Auxiliary Feedwater Pump start. The result could be a water hammer event that concludes with a steam pipe rupture. However, based on the evidence presented above, this sequence of events is also unlikely. The administrative controls provided by the new SLC and existing plant procedures and programs will ensure that there is not more than a minimal increase in the frequency of a steam line break due to the scenario where a section of Main Steam to Auxiliary Equipment System heat trace fails in the de-energized state prior to a Turbine Driven Auxiliary Feedwater Pump start. The heat trace is not required to be functional while the Turbine Driven Auxiliary Feedwater Pump is running.

Based on the discussion above, a minimum of three infrequent events (heat trace failure, instrumentation failure, and Turbine Driven Auxiliary Feedwater Pump auto-start) would have to occur in sequence for the possibility of a steam line break to occur. The heat trace chart recorder and annunciator have separate reliable (but not nuclear safety related) power supplies - which provides diversity with respect to common cause failures. This sequence would have to occur in a twelve hour time period between operator rounds when the chart recorder is checked. For both scenarios, operation of the Auxiliary Feedwater Pump Turbine steam supply piping heat trace and pipe temperature monitoring instrumentation, in conjunction with the administrative controls provided by the new SLC and existing plant procedures and programs, will not result in more than a than a minimal increase in the frequency of occurrence of steam line break or any other accident

previously evaluated in the UFSAR

The steam line break scenario discussed previously is the only postulated event related to the Main Steam to Auxiliary Equipment System heat trace pipe temperature monitoring instrumentation and new SLC that could cause a malfunction of the Turbine Driven Auxiliary Feedwater Pump or any other system, structure or component important to safety. It was concluded that there is not more than a than a minimal increase in the frequency of occurrence of steam line break or any other accident previously evaluated in the UFSAR. Therefore, operation of the heat trace and pipe temperature monitoring instrumentation, in conjunction with the administrative controls provided by the new SLC and existing plant procedures and programs, would not result in more than a minimal increase in the likelihood of a malfunction of the Turbine Driven Auxiliary Feedwater Pump - or any other system, structure or component important to safety. In fact, the Turbine Driven Auxiliary Feedwater Pump would not function properly without an operational heat trace system during stand-by readiness periods. Therefore, implementation of the activities described in this evaluation would actually decrease the likelihood of a malfunction of the Turbine Driven Auxiliary Feedwater Pump.

The potential for an increase in the consequences of an accident would come from inability of the Turbine Driven Auxiliary Feedwater Pump to perform required post accident functions. Per Chapter 15 of the UFSAR, the Auxiliary Feedwater System is required to respond to several design basis events, including: Main Feedwater Line Rupture, Main Steamline Rupture, Small and Large Break LOCAs, Control Rod Ejection, Loss of Normal Feedwater/LOOP, Steam Generator Tube Rupture, Locked Reactor Coolant Pump Rotor and Uncontrolled Single Rod Withdrawal. It was previously shown that the reliability of the Turbine Driven Auxiliary Feedwater Pump will not be more than minimally affected by the operation of the Auxiliary Feedwater Pump Turbine steam supply piping heat trace and pipe temperature monitoring instrumentation, in conjunction with the administrative controls provided by the new SLC and existing plant procedures and programs. The health and safety of the public, with respect to the radiological consequences of the described activities, will not be affected. Therefore, there will not be more than a minimal increase in the consequences of an accident as described in the UFSAR.

Several Catawba accident analyses, including that for Feedwater System Pipe Break, assume a single failure of the Turbine Driven Auxiliary Feedwater Pump. Thus, loss of the Turbine Driven Auxiliary Feedwater Pump has been previously evaluated in the UFSAR. Loss of the Turbine Driven Auxiliary Feedwater Pump as a result of the activities described in this evaluation would not have a different consequence than previously evaluated. Radiological consequences with respect to reactor core damage and off site dose would not change. Therefore, the consequences of a malfunction of the Turbine Driven Auxiliary Feedwater Pump will not change as a result of operation of the Auxiliary Feedwater Pump Turbine steam supply piping heat trace and pipe temperature monitoring instrumentation, in conjunction with the administrative controls provided by the new SLC and existing plant procedures and programs.

Catawba was not analyzed for a steam line rupture in the Auxiliary Building. However, the Auxiliary Feedwater Pump Turbine steam supply piping runs through the Auxiliary Building en route to the turbine. Section 4.3.5 of NEI 96-07 "Guidelines for 10CFR50.59 Implementation" Rev. 1 states: "This criterion (10CFR50 59 Evaluation, Question 5)

deals with creating the possibility for accidents of similar frequency and significance to those already included in the licensing basis for the facility. Thus, accidents that would require multiple independent failures or other circumstances in order to "be created" would not meet this criterion.

Rupture of the Auxiliary Feedwater Pump Turbine steam supply piping due to the conditions presented in this evaluation is not a credible scenario A minimum of three "multiple independent failures" (heat trace failure, instrumentation failure, and Turbine Driven Auxiliary Feedwater Pump auto-start) would have to occur in sequence for the possibility of a steam line break to occur. The heat trace, chart recorder, and annunciator have separate reliable (but not nuclear sfaety related) power supplies - which provides diversity with respect to common cause failures. In addition, this entire sequence would have to occur in a twelve hour time span, between operator rounds, when the chart recorder is inspected. Even if these events occurred in the twelve hour time span, there is no certainty that a pipe rupture would actually occur.

Based on the guidance provided in NEI 96-07, operation of the Auxiliary Feedwater Pump pipe heat trace and temperature monitoring instrumentation, in compliance with the new SLC and other administrative controls, will not create the possibility for an accident of a different type than previously evaluated in the UFSAR.

No malfunctions with a different result than those evaluated in the UFSAR have been identified. Steamline rupture (in containment or doghouse), and loss of the Turbine Driven Auxiliary Feedwater Pump have been previously evaluated in the UFSAR. The likelihood and results of these events will not change as a result of the heat trace related activities discussed in this evaluation. Steamline rupture in the Auxiliary Building has been ruled out as a credible event. Thus, the effect of a steam line rupture on Auxiliary Building equipment does not require evaluation. There are no other credible failure modes or malfunctions associated with operation of the Auxiliary Feedwater Pump Turbine steam supply heat trace system in conjunction with the described SLC and other plant administrative requirements.

Operation of the Auxiliary Feedwater Pump Turbine steam supply heat trace system, in conjunction with the described SLC and other plant administrative requirements, will not affect the design basis limits for the containment vessel, Reactor Coolant System pressure boundary or the fuel cladding. The ability of the Turbine Driven Auxiliary Feedwater Pump and the Auxiliary Feedwater System to remove primary plant decay heat, will not be affected.

The heat trace related activities described in this evaluation do not affect any method of evaluation used in establishing the design bases or in the safety analyses. The activities involve plant equipment and the administrative requirements for operating, testing and maintaining that equipment.

10 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Sections 11.5 1.2 2.1 and 11.5.1.2.2.10

Description: UFSAR Section 11.5.1.2.2.1 "Unit Vent Airborne Monitor" and Section 11.5.1.2.2.10 "Waste Monitor Tank Building Ventilation Monitor" are being changed to revise the description of the Radiation Monitors. There is no actual change to the operation or performance of the monitors in the field. The change is generally an editorial change except for the removal of specifications associated with the Unit Vent Airborne Monitor Sample Pump and the presence of a nozzle for obtaining isokinetic samples of gaseous releases. In order to maintain isokinetic conditions for particulate sampling, the sample flow rate must be varied with relation to changes in the Unit Vent. Therefore specifying a 5 scfm flow rate for the Unit Vent Airborne Monitor is not appropriate Similarly, isokinetics only applies to particulates (refer to ANSI N13 1-1969). Therefore references to isokinetic gas sampling are also not appropriate.

Evaluation: The Radiation Monitoring System is not an an accident initiating system. Therefore a modification of the UFSAR description of the system will have no effect on any accident evaluated in the SAR. A 10CFR50.59 evaluation concluded that the change could be made without prior NRC approval. No Technical Specification changes are required. UFSAR Section 11.5.1.2.2.1 and Section 11.5.1.2.2.10 will be revised.

8 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitments 16 9-7 and SLC 16.9-9.

Description: This Selected Licensee Commitment (SLC) change will revise SLC 16.9-7 "Boration Systems Flowpaths - Shutdown" and 16.9-9 "Boration System Pumps - Shutdown" to delete Note 1 and any reference to the note. In both SLCs, Note 1 states "Since the Boron Dilution Mitigation System (BDMS) is inoperable when the Residual Heat Removal and Safety Injection Pump options are used, the operator must log BDMS inoperable and enter the appropriate action per Technical Specification (TS) 3.9.2." Engineering evaluated whether BDMS could remain operable when a boration flowpath other than the Chemical and Volume Control System Centrifugal Charging Pumps was used. The specific concern is when either a Residual Heat Removal Pump or a Safety Injection Pump is used as the boration pump.

A review of TS 3.3.9 "BDMS" and TS 3.9.2 "Nuclear Instrumentation" indicates that two trains of BDMS are required operable in Modes 3, 4 and 5 (TS 3.3.9) and in Mode 6 (TS 3.9.2). For operability, the following must be performed: channel checks, channel operational tests (COTs), verification that automatic valves move to the correct position and the Reactor Makeup Water Pumps stop upon receipt of the appropriate signal. This means that each of the channels of BDMS must be indicating correctly, provide appropriate alarms, open valves NV-252 and NV-253 on hi alarm, close valves NV-188 and NV-189 on high alarm, and trip the Reactor Makeup Water Pumps on hi alarm. The purpose of each of these actions is to mitigate the consequences of an inadvertent dilution of the reactor coolant.

A review of pertinent references showsthat BDMS is designed to perform the following functions:

1.Detect an increase in the neutron count rate in the reactor core above a pre-set alarm setpoint. Excore detectors are installed to measure the neutron count rate outside the reactor vessel. The BDMS function of detection and providing the appropriate alarms to the operators is not affected by whether the boration flowpath uses the Chemical and Volume Control System, or the Safety Injection System, or the Residual Heat Removal System.

2. Provide an alarm to the operators of an increase in neutron count rate in the reactor core. The detectors feed into installed circuitry that is designed to provide audio and visual alarms if the neutron count rate exceeds a predetermined value. The BDMS function of detection and providing the appropriate alarms to the operators is not affected by whether the boration flow path uses the Chemical and Volume Control System, or the Safety Injection System, or the Residual Heat Removal System.

3. Automatically stop a potential dilution of the Reactor Coolant System at the source. The Boron Recycle System input to the Chemical and Volume Control System makeup system is considered as the source of dilutions into the Reactor Coolant System. BDMS will automatically stop the Reactor Makeup Water Pumps and close the Volume Control Tank outlet valves (NV-188A and NV-189B) whenever it is actuated. This is considered as adequate to stop a dilution of the Reactor Coolant System. UFSAR Chapter 15 "Safety Analysis" states that for makeup water to be added to the Reactor Coolant System at

pressure, at least one charging pump must be running in addition to a Reactor Makeup Water Pump. If a Safety Injection System or Residual Heat Removal System Pump is being used with the Reactor Coolant System at pressure, this does not create a problem since a charging pump is not being used. UFSAR Chapter 15 also states that when the Reactor Coolant System is not at pressure, the rate of addition of unborated makeup water is limited by administratively controlling the output of the Reactor Makeup Water Pumps This limit is not changed when using a Safety Injection System, or the Residual Heat Removal System Pump. For either case above, (Reactor Coolant System pressurized or unpressurized), BDMS will still stop the Reactor Makeup Water Pumps even if a Safety Injection System Pump or the Residual Heat Removal System Pump is being used.

4. Automatically aligns a borated water source for injection into the Reactor Coolant System.
BDMS will automatically align the suction of the Chemical and Volume Control System Centrifugal Charging Pumps to the Refueling Water Storage Tank from the Volume Control Tank. The Volume Control Tank outlet valves (NV-188A and NV-189B) are automatically closed as soon as the Chemical and Volume Control System Pump suction valves from the Refueling Water Storage Tank (NV-252A and NV-253B) are detected as going open whenever BDMS is actuated. The Refueling Water Storage Tank is a borated water source normally kept available to assure that the core remains subcritical and cool. It should be noted that operator action is required to start a Chemical and Volume Control System Pump if it is not already running.

When either a Residual Heat Removal System or Safety Injection System Pump is being used as the boration pump, the Selected Licensee Commitment requires the flowpath from the Refueling Water Storage Tank to the Reactor Coolant System (through the respective pump) to be operable. This meets the intent of BDMS realigning the suction of the Chemical and Volume Control System Pump to the Refueling Water Storage Tank (borated water source). Also if power is not isolated to valves ND-252, NV-253 and NV-188, ND-189, they will reposition even if a Residual Heat Removal System or Safety Injection System Pump is being used as the boration pump. This would have no adverse effect on the boration pump. Note that operator action is required to start a Chemical and Volume Control System Pump, and the same is true for both a Safety Injection System or Residual Heat Removal System Pump. The discharge flowpath, when using a Chemical and Volume Control System Pump, does not get any automatic signal from BDMS. In the case where a Safety Injection System or Residual Heat Removal System Pump is being used, the discharge flowpath to the Reactor Coolant System also gets no automatic signal from BDMS. In the case where a Safety Injection System or Residual Heat Removal System Pump is being used, the discharge flowpath to the Reactor Coolant System also gets no automatic signal from BDMS. The flowpath must be aligned no matter what pump is used as the boration pump.

In order for BDMS to remain operable, it must still be able to perform the functions above when using the Safety Injection System or Residual Heat Removal System. When the Safety Injection System or Residual Heat Removal System is used as a boration flowpath to the Reactor Coolant System, it must be proven that it accomplishes the BDMS functions of isolating a dilution source and is capable of providing borated water to the Reactor Coolant System. The BDMS functions are automatic. With no operator action required. As can be seen from the above discussion, using a Safety Injection

System or Residual Heat Removal System Pump instead of a Chemical and Volume Control System Pump does not prevent BDMS from meeting the intent of its function without operator action. Using a Safety Injection System or Residual Heat Removal System Pump instead of a Chemical and Volume Control System Pump as the boration pump does not make BDMS inoperable. Deleting the note and all references to it in SLC 16.9-7 and SLC 16.9-9 is acceptable.

Evaluation: UFSAR Section 15.4.6.2 discusses purging the unborated water from the piping leading to the Reactor Coolant System Per Safety Analysis, purging the line is a penalty in the analysis. UFSAR Section 15.4.6 1 says that for reactor makeup water to get into the Reactor Coolant System, at least one Chemical and Volume Control System Pump must be running. In the case where either a Residual Heat Removal System Pump or a Safety Injection System Pump is being used , there would be no concern with reactor makeup water getting into the Reactor Coolant System. Also if power is not isolated to the Reactor Makeup Water Pumps, they will trip even if a Residual Heat Removal System or a Safety Injection System Pump is being used as the boration pump. This would have no adverse effect on the boration pump. Therefore using a Residual Heat Removal System Pump or a Safety Injection System Pump would not adversely affect UFSAR Section 15.4 6.

A note will be added to the bases of Selected License Commitment 16.9-7 and Selected License Commitment 16.9-9 to reference the Corrective Action Program report which deleted this note from each of the Selected License Commitments. A 10CFR50.59 evaluation concluded that this change could be made without prior NRC approval. No Technical Specification changes are required