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

SUBJECT: Duke Energy Corporation
Catawba Nuclear Station Unit 1 and Unit 2
Docket Numbers 50-413 and 50-414
2001 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 2001. 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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Units 1 and 2

2001 10CFR50.59 Report

April 1, 2002

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 2001. The entries are organized by the type of activity being evaluated in the following order:

52 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-10974, Generate flow diagrams for Nuclear Service Water System Plastic Drain Piping and Ground Drainage System Plastic Discharge Piping

Description: Minor modification CE-10974 generates flow diagram changes to show existing Nuclear Service Water System plastic drain piping and Groundwater Drainage System plastic discharge piping. The Nuclear Service Water System plastic drain piping runs from the Nuclear Service Water System Header drain valves on Auxiliary Building elevation 577' to the Groundwater Drainage System Sump. The piping is used for draining the Nuclear Service Water System header for maintenance purposes when the associated portion of the header is isolated and out of service. The Groundwater Drainage System plastic piping runs from Groundwater Drainage System Sump C up and out of the Auxiliary Service Building and discharges into the yard. This modification showed that the non-nuclear related piping and associated components satisfy applicable design requirements. This modification made minor changes to enhance the existing piping isometrics for the plastic piping. The provision for connecting the Nuclear Service Water System plastic drain piping to the Interior Fire Protection System was eliminated by this modification.

The function or operation of the Nuclear Service Water System, Groundwater Drainage System, and the Interior Fire Protection System was not affected by this modification. These systems will continue to function as described in the UFSAR and the System Design Basis Specifications.

Evaluation: There are no unreviewed safety questions as a result of this modification. The modification will have no effect on any of the accidents analyzed in the UFSAR. No Technical Specification changes are required. UFSAR Figures 9-23, 9-28, 9-139, 9-140 and 9-195 (system flow drawings) will be revised.

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

Title: Minor Modification CE-60388, Revise documents associated with the Low Pressure Service Water System Intake Screen Backwash System

Description: Minor Modification CE-60388 will revise drawings and documents associated with the Low Pressure Service Water System Intake Screen Backwash System to reflect "as-built" conditions. Also included in the modificaton are removal of control board switches, cables, heat tracing, valves, piping and instrumentation associated with the previously removed Low Pressure Service Water intake screen backwash pump. The Low Pressure Service Water intake screen backwash pump which supplied water to the intake screen for backwash was previously removed. The Low Pressure Service Water Pumps will provide the water to the intake screens for the backwash operation.

Evaluation: Removal of the Low Pressure Service Water intake screen backwash pump and associated components will have no effect on any of the accidents analyzed in the UFSAR. There are no unreviewed safety questions associated with this minor modification. No Technical Specification changes are required. UFSAR Section 9.2.8.2, Figure 9-63, and Figure 2-44 (Part 1 of 3) will be revised.

53 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-61172, Abandon in place 0RNPS6000 and 0RNPS6040 associated with the Control Room Area Chillers

Description: Minor Modification CE-61172 will disable the Nuclear Service Water differential pressure flow interlocks with each of the Control Room Area Chillers. These Nuclear Service Water differential pressure switches, 0RNPS6000 and 0RNPS6040 will be abandoned in place. The interlock with the Control Room Area Chillers will be removed. The two red indicating lights located on control board 1MC3 (Nuclear Service Water Train A and B Water Flow) that receive their input from 0RNPS6000, 0RNPS6040 will be removed and their holes plugged. The "Condenser Low Water Flow" indicating lights located on the Control Room Area Chillers control panel will be removed.

The Control Room HVAC chillers are nuclear safety related. These chillers are designed to provide chilled water to cool the Control Room and Control Room Area during all plant conditions. The Nuclear Service Water differential pressure switches function to give a permissive for the chiller to start and run only if Nuclear Service Water System pressure is established across the chiller condenser and the Nuclear Service Water flow control valve. If the pressure differential across the condenser and control valve is less than five psi, the chiller will not start or run. These switches do not work properly due to the way in which they are connected to the piping. The switches are connected across the control valves so that when the control valve is closed and there is no flow through the condenser the switch still senses a high differential pressure. The only time this switch will protect the chiller is if the chiller is in operation and the Nuclear Service Water flow to the condenser is lost. If this were to happen the chiller would trip on high condenser pressure (the pressure switches are redundant trips).

The Nuclear Service Water differential pressure switches have proven to be a maintenance expense and an operating risk, thus the need to remove their interlocks with the Control Room Area Chillers. Additionally the need for future maintenance resources will be reduced by deleting the two Nuclear Service Water differential pressure switches and the corresponding Work Orders used to calibrate the switches.

Evaluation: The removal of the Nuclear Service Water differential pressure flow interlocks and associated flow indication will not increase the probability of an accident evaluated in the SAR, the Control Room Area Chillers will continue to perform their functions during an accident, thus the consequences of an accident evaluated in the SAR will not increase. Onsite and offsite dose analysis will not be affected. No new credible accidents will be created, thus the possibility for an accident of a different type than those evaluated in the SAR will not increase. Reliability will increase by reducing the number of interlocks while still maintaining equipment protection, thus the probability of a malfunction of equipment important to safety evaluated in the SAR will not increase. The Control Room Area Chillers will still trip on high condenser pressure when condenser flow is lost, thus the consequences of a malfunction of equipment important to safety evaluated in the SAR will not be increased. No unanalyzed malfunction will be created, thus the possibility for a malfunction of a different type than any evaluated in the SAR will not be created. No Technical Specification margins of safety will be affected. Control Room temperature will still be controlled and maintained within required limits, thus the margin of safety as defined in the basis for any Technical Specification will not be reduced.

There are no Unreviewed Safety Questions associated with this modification. No Technical Specification changes are required. A SAR revision is required to revise the Nuclear Service Water System flow diagrams (UFSAR Figure 9-27 and 9-31) to show the switches as "abandoned in place'. UFSAR Section 7.6.13.1 will be revised.

49 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-61493, Re-rate Instrument Air System Compressor D

Description: Minor Modification CE-61493 will "rerate" Instrument Air Compressor D. Instrument Air Compressor D will be rerated by Ingersoll-Rand so that it will have the same design specification performance capacity as Instrument Air Compressors E and F. This will involve replacing the air end of the compressor which includes internal parts such as the impeller and diffuser. Also, Instrument Air System Compressor D discharge relief valve 1VI-497 will be replaced due to the higher design pressure that results from the modification.

This modification will increase the operational reliability of the Instrument Air System. This will be accomplished by modifying Compressor D so that its design specifications are the same as Compressors E and F. This will allow all three Instrument Air Compressors to operate in the most efficient and reliable configuration. The required design specifications for the Instrument Air System are maintained. The Instrument Air System will continue to operate as currently described in the UFSAR. The new relief valve has been evaluated to ensure it is designed for the new design parameters of the re-rated Compressor D.

Evaluation: There are no unreviewed safety questions associated with this modification. Although the loss of the Instrument Air System can initiate a Turbine/Reactor trip, this modification will improve the reliability of the system. All safety related accident mitigation equipment can perform its intended function independent of the Instrument Air System. No new failure mode of the Instrument Air System is introduced due to this modification. No Technical Specification changes are required. UFSAR Figures 9-70 and 9-71 (piping flow drawings) will be revised.

14 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-61502, Replace Obselete Dew Point Monitor System

Description: Minor Modification CE-61502 will replace the existing obsolete dew point monitoring system, which consists of a remote sensor mounted at the ten meter elevation on the MET Tower and control unit located in the MET building, with a new dew point monitoring system. No changes are required to the plant (i.e.: recorders, Operator Aid Computer, etc.), because the new dew point monitoring system will have the same range as the existing dew point monitoring, -30 degrees C to +30 degrees C.

In addition, drain vents will be installed in the terminal box at the base of the MET Tower to allow the elimination of any accumulation of moisture from the box.

Evaluation: The meteorological measurement system is used to aid in the assessment of the maximum potential annual radiation dose to the public from the routine release of radioactive gaseous effluent, the potential dispersion of radioactive materials from and the radiological consequences of a spectrum of accidents, and non-radiological environmental effects such as fogging and icing from the cooling towers. The dew point monitoring system is used in the assessment of the last item. The existing dew point monitoring system is obsolete and is not functioning properly.

This modification will replace the existing obsolete dew point monitoring system with a new dew point monitoring system. The new equipment will meet or exceed the requirements of the meteorological design specification. The new dew point monitoring system will interface with the existing meteorological system the same as the existing dew point monitoring system (i.e.: same ranges, no changes to recorder or Operator Aid Computer inputs, etc.). There are no unreviewed safety questions associated with this minor modification. No Technical Specification changes are required. UFSAR Section 2.3.3.2 will be revised to correctly reflect the requirements for the dew point monitoring system.

22 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CE-61507, Document Removal of the Liquid Waste Recycle System Waste Evaporator Concentrates Pumps

Description: Minor Modification CE-61507 will document removal of the Liquid Waste Recycle System Waste Evaporator Concentrates Pumps. The Waste Evaporator Package has never been used due to the addition of the Monitor Tank Building.

Evaluation: The removal of the Concentrates Pumps will have no effect on any of the accidents analyzed in the UFSAR. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. UFSAR Figures 9-41, 9-43, 11-5, 11-6, 11-7, 11-9 and 11-33 will be revised.

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

Title: Minor Modification CE-61515, Replace Main Feedwater System piping reflective insulation at Steam Generator 2B with fiberglass blanket insulation

Description: Minor Modification CE-61515 will replace the "mirror insulation" on the 18" Main Feedwater System piping to Steam Generator 2B with nuclear grade fiberglass blankets. The affected section of piping will be from the Steam Generator 2B inlet nozzle back through the 18" x 16" reducer. The fiberglass blankets will be five inches thick with stainless steel jacketing.

Currently, elevated temperatures in the "B" quadrant are affecting the operation of Lower Containment Ventilation Unit (LCVU) 2B. One likely source of heat load may be the proximity of the 2B Main Feedwater System piping to the "B" fan room window. This piping is insulated with mirror or reflective insulation that has a surface temperature of 155 degrees F. at 100% power. Although the mirror insulation is not damaged and is meeting its intended design function, a replacement with blanket insulation will likely reduce the heat load in the affected area due to the better insulating qualities of the fiberglass insulation. The nuclear grade fiberglass blanket insulation used in the Reactor Building has been evaluated and determined to be not nuclear safety related. However, the weight of the insulation is considered in the piping stress models for all systems analyzed by stress analysis. UFSAR Section 6.3.4.1 documents the use of blanket insulation in the Reactor Building.

The containment sump screen has been evaluated for the clogging potential of qualified blanket insulation used as an option to the original reflective insulation. In order to minimize the sump screen loading due to paint chips or blanket insulation material which has been affected by the LOCA or Steam Line Break, the bottom row of pipe sleeves closest to the floor have been closed. The remaining open sleeves at higher elevations are sufficient to allow the free movement of water from the area inside the crane wall (where all high energy pipe is located) to the tunnel area outside the crane wall, where the sump recirculation screen is located. The Reactor Building Design Basis documents the use of blanket insulation. Nuclear Grade fiberglass blanket insulation is currently being used at Catawba. Several modifications have been implemented to replace the reflective, or mirror insulation with these blankets. The Unit 1 Steam Generator Replacement Project (Modification CN-19230/00) installed this blanket insulation on the Steam Generators and the associated feedwater piping.

Evaluation: Blanket insulation is suitable for use on the Main Feedwater System piping and will not affect the design, stress analysis or integrity of the affected piping. The blanket insulation meets the design insulated heat transfer rate requirement and actually insulates better than mirror insulation. Calculations have analyzed the effect that blanket insulation debris could have on the Containment Recirculation Sump and ECCS pump operation. These calculations conclude that blanket insulation is suitable for use on piping and components in containment inside the crane wall. This modification does not involve an unreviewed safety question. No Technical Specification change is required. No UFSAR changes are required.

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

Title: Minor Modification CE-61532, Restore Valve 2CA-6 to operable status

Description: Minor Modification CE-61532 reverses the changes to valve 2CA-6 made by Minor Modification CE-61397 which changed the position of valve 2CA-6 so that the valve is normally closed with the associated motor operator breaker in the "off position. After implementation of modification CE-61532, valve 2CA-6 will be normally open.

The changes to the Auxiliary Feedwater and Condensate Storage Systems made by CE-61532 were needed due to the problems with the Auxiliary Feedwater System Condensate Storage Tank (CACST) supply identified in 1997. Currently, valve 2CA-6 is closed with power removed to isolate the CACST from the Auxiliary Feedwater pumps' supply piping. This valve is closed to prevent the introduction of air into the Auxiliary Feedwater System pump suction piping. Due to the piping configuration of the Auxiliary Feedwater and Condensate Storage Systems, during certain plant conditions and upon depletion of the CACST, air could be drawn into the Auxiliary Feedwater System suction piping which could result in Auxiliary Feedwater pump damage. Minor modification CE-61532 and Nuclear Station Modification CN-21401/00 resolve the air entrainment issue and restore the CACST to service. Nuclear Station Modification CN-50477 will add a dedicated Unit 2 CACST.

Evaluation: NSM CN-21401/00 reroutes and enlarges the Upper Surge Tank (UST) to CACST junction piping and resolves the concern for air entrainment as described in the paragraph below. Minor Modification CE-61532 can not be implemented until Nuclear Station Modifications CN-21401/00 and CN-50477/00 are completed.

The air entrainment issue resolved by NSM CN-21401/00 is as follows:

On May 18, 1997 concerns with vortex formation in the CACST and USTs were identified that could lead to the introduction of air into the suction piping of the Auxiliary Feedwater pumps, potentially disabling the pumps. An Operability Evaluation was performed, with support from the Auxiliary Feedwater pump vendor, which concluded that vortex formation is not an operability concern. During the process of evaluating the vortex concern, a separate mechanism was identified by which air could potentially enter the Auxiliary Feedwater suction piping. This mechanism involved the depletion of the CACST and the failure of valve CA-6 (CACST to Auxiliary Feedwater Pump Isolation Valve) to automatically close on a low CACST level. In this situation, the USTs would supply the Auxiliary Feedwater pumps. However, if condenser vacuum is not broken, the relative elevation head of the UST's with respect to the junction of the Auxiliary Feedwater supply piping from the CACST and UST's is not sufficient to maintain the pressure at this junction above atmospheric pressure over the full range of possible Auxiliary Feedwater Syatem flow rates. Also, due to the elevation difference between this junction and the pressure switches that activate the automatic swapover to the assured Auxiliary Feedwater System suction sources of the Nuclear Service Water System and the setpoint of these pressure switches, the swapover is not assured if the CACST - UST junction pressure is less then atmospheric over the full range of possible Auxiliary Feedwater System flows. This could lead to the introduction of air into the Auxiliary Feedwater System suction piping and possibly into the Auxiliary Feedwater pumps from the depleted and unisolated CACST, potentially disabling the pumps.

Minor modification CE-61532 does not add or delete any automatic or manual safety related feature of the Auxiliary Feedwater System, nor does it convert an automatic safety related feature to manual or vice versa. The modification does not introduce an unwanted or previously unreviewed system interaction, but instead eliminates such an interaction. This modification does not alter the QA condition, or seismic or environmental qualification of any component in the Auxiliary Feedwater System as the CACST is a non-safety, non-seismic tank. No adverse effects on the safety related function of the Auxiliary Feedwater System or any interfacing systems are created by this activity. The Condensate Storage System will continue to provide the Technical Specification required condensate inventory of approximately 225,000 gallons. The Nuclear Service Water System will continue to be the safety related supply for the Auxiliary Feedwater System and this function is not affected. No new failure modes are created by this modification.

No unreviewed safety questions are created as a result of CE-61532 which unisolates valve 2CA-6 and thus restores the functionality of the CACST. Changes to Technical Specification 3.7.6 are not required; however, the bases for this Technical Specification will be revised to reflect the restored CACST. This modification does result in the plant configuration being different from that described in the UFSAR (specifically Section 10.4.9.2 and Appendix 10 Chapter 10 Tables and Figures). The affected sections and figures will be revised to show the new configuration of the Auxiliary Feedwater and Condensate Storage Systems.

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

Title: Minor Modification CE-61547, Add a Lifting Device in the Unit 2 Annulus for Flanges for Steel Containment Vessel Penetrations MK M371 and MK394

Description: A personnel safety concern was identified associated with the removal and reinstallation of spare containment penetration flanges M394 and M371. These penetrations are for access of temporary electrical power, compressed air supply and ice condenser hoses during refueling outages. Modification CE-61547 will provide a lifting device (davit assembly), attached to each penetration sleeve, on the Annulus side, to allow safe removal and replacement of the penetration blind flange.

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

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

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

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

50 Type: Minor Modification **Unit:** 0

Title: Minor Modification CE-61568, Increase the automatic start and fast start setpoints for the third standby instrument air system compressor

Description: Modification CE-61568 will increase the automatic start setpoints for the third Instrument Air System standby compressor, as an early mitigation activity for the case of Instrument Air System pressure degradation. This activity is intended to occur prior to or concurrent with the initiation of the Loss of Instrument Air System Abnormal Operations Procedure (AP/O/A/5500/22). This early mitigation response activity will be achieved by increasing the Low Pressure Alarm Setpoint and Low Pressure Emergency Alarm Setpoint from 92 psig and 90 psig, respectively, to 96 psig and 94 psig respectively.

The Instrument Air System operating pressure (setpoint) is 100 psig, measured at the compressor discharge location. The Instrument Air System air demand requires two of the three compressors to operate continuously to meet demand and maintain the 100 psig setpoint. For the case of a rapid decay in system pressure, such as an operating Instrument Air System compressor trip or a pipe rupture, the Centac Energy Management (CEM) Computer is designed to start the third (standby) compressor. This start function is activated by the Low Pressure Alarm Setpoint. If the pressure decay is extreme, the fast automatic response Low Pressure Emergency Alarm Setpoint may be activated. Currently, these setpoints are 92 psig and 90 psig, respectively. The setpoints have been determined to be non conservative when compared to the setpoints of the Loss of Instrument Air System Abnormal Operations Procedure (AP/O/A/5500/22). That is, a degrading pressure condition may occur, requiring the initiation of the Loss of Instrument Air System Abnormal Procedure. Initiation of the Loss of Instrument Air System Abnormal Procedure may be necessary prior to the automatic start of the third (standby) Instrument Air System compressor. It is not the intent of the Loss of Instrument Air System Abnormal Procedure to initiate manual mitigating activities without utilizing this automatic mitigating function.

The Loss of Instrument Air System Abnormal Procedure setpoint of 85 psig requires specific manual Operator actions to commence for the degraded condition. The location of this instrumentation measures system pressure downstream of the Instrument Air System dryers. This location of the Control Room indication maintains a 7-12 psig differential pressure lower than the compressor indication location. This variation of 7-12 psig is due to the piping and the Instrument Air System air filters/dryer skid differential pressure that occurs between preventative maintenance work.

Therefore under normal conditions the compressors are maintaining 100 psig and the normal control room indication will display 93 to 88 psig. Currently the pressure decay at the compressors would decrease to 92 psig prior to initiating the autostart function of the third compressor. The equivalent Control Room indication would display 85 to 80 psig. This range is within the limit of the Loss of Instrument Air System Abnormal Procedure. Thus, the Operators at the Controls would probably be required to initiate the Loss of Instrument Air System Abnormal Procedure, prior to the current automatic start of the third Instrument Air System compressor.

The change from this modification will not affect the description of the Instrument Air

System, as stated in the Design Basis, the UFSAR Section 9.3.1, or affect how the system is currently being operated.

Evaluation: The Instrument Air System provides the compression, filtration, storage, and delivery of instrument quality control air to all air operated instrumentation and valves for plant operation. This modification will increase the recovery time for Control Room Operators in the event of a degrading Instrument Air System. The system is a non-safety-related system with the exception of the Containment Penetrations. The system is not required to achieve a safe reactor shutdown or to mitigate the consequences of an accident. The original design of the system remains unaffected by this modification. The Instrument Air System will continue to be operated as intended in accordance with manufacturer's specifications, and within the intent of the UFSAR. It will not have an effect on plant safety or any accidents evaluated in the plant accident analyses. No Technical Specification changes are required. No UFSAR changes are required.

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

Title: Minor Modification CE-61593, Delete unused RTDs from the Ice Condenser Temperature Monitoring System and abandon in place the Multi Point Selector Switch Assembly with Indicator Panel (2NFP6070)

Description: Minor Modification CE-61593 will delete unused, non Technical Specification Related Resistance Temperature Detectors (RTDs), along with the associated wiring and conduit, from the Ice Condenser Temperature Monitoring Subsystem. The RTDs and associated wiring will be deleted back to their respective terminal boxes. The cables from the terminal boxes to 2NFP6070 will be taken to ground in the terminal boxes, and remain terminated at 2NFP6070. 2NFP6070 will be abandoned in place to avoid unnecessary work in lower containment. The equipment to be deleted is comprised of three groups of RTDs identified as Floor Cooling, Wall Panel Mounted and Wear Slab Mounted RTDs.

The Floor Cooling, Wall Panel Mounted and Wear Slab Mounted RTDs, which comprise the non-Technical Specification related RTDs of the Ice Condenser Temperature Monitoring Subsystem, do not serve a safety related function, nor interact with any safety related components or equipment or any equipment important to safety. This portion of the Ice Condenser Temperature Monitoring subsystem is no longer utilized by any plant personal, yet maintenance on this part of the system is still required. Removal of this equipment will eliminate the need for maintenance. Technical Specification related RTDs will remain active to provide data to the control room where ice condenser temperatures are monitored for anomalies.

Evaluation: This modification has no effect on any of the accidents analyzed in the UFSAR. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. UFSAR changes are required for UFSAR Table 6-127 and UFSAR Figure 6-176

89 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-61594, 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.

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.

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

Title: Minor Modification CE-61595, 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-61595 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/2/A/6100/002. The P-12 interlock will be bypassed in the Auxiliary Safeguards Cabinets to disable the interlock when appropriate pressure and temperature conditions are met during Unit 2 cooldown in Mode 4. Technical Specification 3.3.2 requires that the interlock be operable during Modes 1, 2, and 3. This interlock may be bypassed when the unit is in Mode 4 since it is no longer required by Technical Specifications. The condenser steam dumps are controlled using the Steam Pressure Controller before and after the P-12 interlock is bypassed. This controller can be operated in auto with a steam pressure setpoint or in manual with a pushbutton demand signal. This procedure reduces the amount of time the Residual Heat Removal System is needed to operate during unit cooldown by performing an extended cooldown using condenser dump valves at lower temperatures. This method of cooldown is expected to reduce the amount of crud precipitated upon start of the Residual Heat Removal System and lower general area dose rates during shutdown.

The changes effected by this modification are:

1. Install a two position key operated switch in Auxiliary Safeguards Cabinets 2AUXSFGA and 2AUXSFGB 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 2 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/2/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.

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.

There are no Unreviewed Safety Questions associated with this modification. 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.

19 Type: Minor Modification **Unit:** 1

Title: Minor Modification CE-61600, Replacement of Temperature Indicators on the Hydrogen Recombiner Temperature Monitoring Panels

Description: The heater-recombination section of the hydrogen recombiners contain four banks of heaters. Heater bank #3 has three Type K (chromel-alumel) thermocouples imbedded in the heater sheaths. These thermocouples are provided to verify heater operation and to indicate plate temperature for Technical Specification performance testing. Technical Specification Surveillance Requirement 3.6.7.1 requires that a functional test be performed for each hydrogen recombiner every 18 months.

Per the Bases for Surveillance Requirement 3.6.7.1:

"Performance of a system functional test for each hydrogen recombiner ensures the recombiners are operational and can attain and sustain the temperature necessary or hydrogen recombination. In particular, this Surveillance Requirement verifies that the minimum heater sheath temperature increases to >/= 700 degrees F. in </= 90 minutes. After reaching 700 degrees F, the power is increased to maximum power for approximately two minutes and power is verified to >/= 60kW.

Three thermocouples are installed in each hydrogen recombiner for performance monitoring. Originally the Hydrogen Recombiner Control Panel was supplied with a temperature indicator to monitor the thermocouples. However, when the recombiners were installed, no Type K thermocouple penetrations were available.

Since no Type K thermocouple penetrations were available, the thermocouple leads needed to be converted to copper leads; however, connecting thermocouple leads to copper leads results in the creation of an additional junction (one in addition to the hot junction, which is monitoring the temperature of interest). Because each junction of dissimilar metal associated with a thermocouple produces a voltage, the junction temperatures need to be taken into account to determine the correct temperature being measured by the thermocouple itself.

To prevent an unknown junction at a standard electrical penetration, a heated reference junction box was added inside containment. The heated reference junction box allows the conversion of thermocouple wire to copper wire at a known temperature, which can be accounted for in determining the correct temperature monitored by the thermocouple. An RTD was provided in each reference junction box to provide constant temperature indication of the reference junction box temperature.

To provide temperature indication, two Hydrogen Recombiner Heater Temperature Monitoring Panels (2ELCP0299 and 2ELCP0300) were installed. Two temperature indicators were installed on each panel. One indicator displays the temperature of the heated reference junction box, as measured by the RTD. The other indicator displays the temperature of one of the three thermocouples, as selected by a selector switch on the panel.

Although the temperature of the junction box is displayed on the panel, the indication does not provide live compensation of the thermocouple indication. The thermocouple indication is compensated for based on a fixed temperature.

The previously installed temperature indicators are obsolete; therefore, this modification will replace the existing obsolete indicators. A single paperless recorder will be used in place of the two temperature indications. Since the recorder can accept both thermocouple and RTD inputs, the recorder will be able to display all four temperature readings. In addition, the RTD monitoring the junction box temperature will be able to provide live compensation of the thermocouple readings. The live compensation will be able to account for any change in junction box temperature; thereby, making the thermocouple indication more accurate. Also, because all the three thermocouples can be directly input to the recorder, the existing selector switch is no longer required, and will be removed.

In addition, documentation which shows that temperature indication is located on the Hydrogen Recombiner Control Panel will be updated to show the temperature indication as removed.

Evaluation: This modification will replace the existing temperature indicators associated with monitoring the heater performance in the Hydrogen Recombiners with a paperless recorder on each Hydrogen Recombiner Heater Temperature Monitoring Panel.

UFSAR Sections 6.2.5.2.1 and 6.2.5.5 discuss the thermocouples installed on the Hydrogen Recombiners, which are used during surveillance testing of the Hydrogen Recombiners to demonstrate operability of the recombiners. As stated in UFSAR Section 6.2.5.5, the thermocouples (and therefore the temperature indication provided by the thermocouples) are provided for convenience in testing and periodic checkout of the recombiners, but are not necessary to assure proper operation of the recombiners. " The indication is only used to verify the operability of the recombiners prior to an accident, to ensure that the recombiners will function after an accident. The temperature indication is not required for the proper operation of the recombiners. The temperature indication is not nuclear safety related and is not required to function during normal or accident conditions.

UFSAR Section 6.2.5.2.1 discusses the temperature indication provided by the thermocouples; however, it incorrectly states that the indication is provided on the Hydrogen Recombiner Control Panel. The indication is actually provided on the Hydrogen Recombiner Heater Temperature Monitoring Panel; therefore, Section 6.2.5.2.1 needs to be revised to reflect the actual readout location.

There are no unreviewed saftey questions associated with this modification. No Technical Specification changes are required. UFSAR Section 6.2.5.2.1 will be revised.

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

Title: Minor Modification CE-61601, Replacement of Temperature Indicators on the Hydrogen Recombiner Temperature Monitoring Panels

Description: The heater-recombination section of the hydrogen recombiners contain four banks of heaters. Heater bank #3 has three Type K (chromel-alumel) thermocouples imbedded in the heater sheaths. These thermocouples are provided to verify heater operation and to indicate plate temperature for Technical Specification performance testing. Technical Specification Surveillance Requirement 3.6.7.1 requires that a functional test be performed for each hydrogen recombiner every 18 months.

Per the Bases for Surveillance Requirement 3.6.7. 1:

"Performance of a system functional test for each hydrogen recombiner ensures the recombiners are operational and can attain and sustain the temperature necessary or hydrogen recombination. In particular, this Surveillance Requirement verifies that the minimum heater sheath temperature increases to >/= 700 degrees F. in </= 90 minutes. After reaching 700 degrees F, the power is increased to maximum power for approximately two minutes and power is verified to be >/= 60kW.

Three thermocouples are installed in each hydrogen recombiner for performance monitoring. Originally the Hydrogen Recombiner Control Panel was supplied with a temperature indicator to monitor the thermocouples. However, when the recombiners were installed, no Type K thermocouple penetrations were available.

Since no Type K thermocouple penetrations were available, the thermocouple leads needed to be converted to copper leads; however, connecting thermocouple leads to copper leads results in the creation of an additional junction (one in addition to the hot junction, which is monitoring the temperature of interest). Because each junction of dissimilar metal associated with a thermocouple produces a voltage, the junction temperatures need to be taken into account to determine the correct temperature being measured by the thermocouple itself.

To prevent an unknown junction at a standard electrical penetration, a heated reference junction box was added inside containment. The heated reference junction box allows the conversion of thermocouple wire to copper wire at a known temperature, which can be accounted for in determining the correct temperature monitored by the thermocouple. An RTD was provided in each reference junction box to provide constant temperature indication of the reference junction box temperature.

To provide temperature indication, two Hydrogen Recombiner Heater Temperature Monitoring Panels (2ELCP0299 and 2ELCP0300) were installed. Two temperature indicators were installed on each panel. One indicator displays the temperature of the heated reference junction box, as measured by the RTD. The other indicator displays the temperature of one of the three thermocouples, as selected by a selector switch on the panel.

Although the temperature of the junction box is displayed on the panel, the indication does not provide live compensation of the thermocouple indication. The thermocouple indication is compensated for based on a fixed temperature.

The previously installed temperature indicators are obsolete; therefore, this modification will replace the existing obsolete indicators. A single paperless recorder will be used in place of the two temperature indications. Since the recorder can accept both thermocouple and RTD inputs, the recorder will be able to display all four temperature readings. In addition, the RTD monitoring the junction box temperature will be able to provide live compensation of the thermocouple readings. The live compensation will be able to account for any change in junction box temperature; thereby, making the thermocouple indication more accurate. Also, because all the three thermocouples can be directly input to the recorder, the existing selector switch is no longer required, and will be removed.

In addition, documentation which shows that temperature indication is located on the Hydrogen Recombiner Control Panel will be updated to show the temperature indication as removed.

Evaluation: This modification will replace the existing temperature indicators associated with monitoring the heater performance in the Hydrogen Recombiners with a paperless recorder on each Hydrogen Recombiner Heater Temperature Monitoring Panel.

UFSAR Sections 6.2.5.2.1 and 6.2.5.5 discuss the thermocouples installed on the Hydrogen Recombiners, which are used during surveillance testing of the Hydrogen Recombiners to demonstrate operability of the recombiners. As stated in UFSAR Section 6.2.5.5, the thermocouples (and therefore the temperature indication provided by the thermocouples) are provided for convenience in testing and periodic checkout of the recombiners, but are not necessary to assure proper operation of the recombiners." The indication is only used to verify the operability of the recombiners prior to an accident, to ensure that the recombiners will function after an accident. The temperature indication is not required for the proper operation of the recombiners. The temperature indication is not Nuclear Safety Related and is not required to function during normal or accident conditions.

UFSAR Section 6.2.5.2.1 discusses the temperature indication provided by the thermocouples; however, it incorrectly states that the indication is provided on the Hydrogen Recombiner Control Panel. The indication is actually provided on the Hydrogen Recombiner Heater Temperature Monitoring Panel; therefore, Section 6.2.5.2.1 needs to be revised to reflect the actual readout location.

There are no unreviewed saftey questions associated with this modification. No Technical Specification changes are required. UFSAR Section 6.2.5.2.1 will be revised.

110 Type: Minor Modification **Unit:** 1

Title: Minor Modification CE-61606, Relocate impulse line tap for RHR Pump discharge pressure instruments 1NDPT5080 and 1NDPT5090 downstream of the pump's discharge check valve to provide Control Room and OAC idication downstream of the check valves

Description: Both Unit 1 and Unit 2 have experienced intermittent Cold Leg Accumulator leakage. This leakage causes pressurization of the Residual Heat Removal Pump discharge headers. Lack of Control Room indication for Residual Heat Removal Pump discharge pressure downstream of each Residual Heat Removal Pump's discharge check valve has hindered monitoring and maintenance activities associated with pressurization of this piping. This has resulted in the need for Operations personnel to check Residual Heat Removal header pressure at local gauges 1NDPG5260 and 1NDPG5270 on operator rounds. It was determined that it was desirable to have Residual Heat Removal Pump discharge pressure downstream of each pump's check valve indicated in the Control Room and input to the Operator Aid Computer. This will be accomplished by relocating the instrument impulse line taps for instruments 1NDPT5080 and 1NDPT5090 downstream of each pump's check valve.

Evaluation: This modification only affects the nuclear safety related instrument impulse lines for instruments 1NDPG5260 and 1NDPG5270. The modification will not affect the operation or function of the Residual Heat Removal System during any phase of normal or accident mitigation operation. This instrumentation does not perform a nuclear safety related function. Having a Control Room indication and an Operator Aid Computer indication of the Residual Heat Removal discharge pressure downstream of the pump discharge check valve will relieve Operations personnel from having to manually log the pressure when the Cold Leg Accumulators are leaking. The affected portions of the Residual Heat Removal System will continue to function as described in the SAR. A 10CFR50.59 evaluation determined that this change could be made without prior NRC approval. No Technical Specification changes are required. UFSAR Figures 5-17 and 5-18 (system flow drawings) will be revised.

39 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-61608, Abandon Containment Floor and Equipment Discharge Monitor

Description: Minor modification CE-61608 will affect the Liquid Radwaste System by abandoning flow element 1WLFE6460, flow transmitter 1WLFT6460, and square root extractor 1WLSR6460 and removing a totalizer and a receiver gauge 1WLP6460 on the main control board. The square root extractor is being added to the Instrument Detail drawing CN-1499-WL.42-00 because it was omitted when the drawing was created, and it will then be shown as "abandoned". An editorial change is being made to an electrical elementary diagram to show the power wiring. The modification will also "void" all abandoned circuits associated with the instrumentation and remove the panel wiring to the termination blocks.

Evaluation: The Containment Floor/Equipment Discharge Flow loop instrumentation is not nuclear safety related. The flow instrumentation is not currently used in any Operations procedures. Therefore, abandoning flow element 1WLFE6460, flow transmitter 1WLFT6460, and square root extractor 1WLSR6460 and removal of a totalizer and receiver gauge 1WLP6460 will not adversely affect plant operation.

Since UFSAR Figure 11-15 shows the flow instrumentation and it is mentioned in Section 11.2.2.7.2.2, changes to the UFSAR are required for this modification. However, since the instrumentation is not providing any signals to any plant control systems, the abandonment/removal of this instrumentation will not adversely affect the operation of the plant during Normal, Abnormal or Accident conditions. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required.

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

Title: Minor Modification CE-61609, Abandon Containment Floor and Equipment Discharge Monitor

Description: Minor modification CE-61609 will affect the Liquid Radwaste System by abandoning flow element 2WLFE6460, flow transmitter 2WLFT6460, and square root extractor 2WLSR6460 and removing a totalizer and a receiver gauge 2WLP6460 on the main control board. An editorial change is being made to an electrical elementary diagram to show the power wiring. The modification will also "void" or spare all circuits associated with the instrumentation and remove the panel wiring to the termination blocks.

Evaluation: The Containment Floor/Equipment Discharge Flow loop instrumentation is not nuclear safety related. The flow instrumentation is not currently used in any Operations procedures. Therefore, abandoning flow element 2WLFE6460, flow transmitter 2WLFT6460, and square root extractor 2WLSR6460 and removal of a totalizer and receiver gauge 2WLP6460 will not adversely affect plant operation.

Since UFSAR Figure 11-22 shows the flow instrumentation and it is mentioned in Section 11.2.2.7.2.2, changes to the UFSAR are required for this modification. However, since the instrumentation is not providing any signals to any plant control systems, the abandonment/removal of this instrumentation will not adversely affect the operation of the plant during Normal, Abnormal or Accident conditions. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required.

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

Title: Minor Modification CE-61627, Replace the existing 2A Chemical and Volume Control System Centrifugal Charging Pump seals with new third generation seals and add a pressure gauge to monitor and trend the pump balance drum pressure

Description: This modification replaces the existing second generation Chemical and Volume Control System 2A Centrifugal Charging Pump mechanical seals with third generation seals supplied by the pump OEM, Ingersoll Dresser Pumps (IDP). The current inboard and outboard seals used on the pumps do not provide adequate service life, involve a significant risk of unrepairable shaft damage during maintenance, and are not the configurations on which active design improvements are made. IDP has shown this seal upgrade and conversion as suitable and appropriate for use with the Chemical and Volume Control System Emergency Core Cooling System (ECCS) pumps.

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

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

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

The Chemical and Volume Control System Centrifugal Charging Pump seal replacement does not change the operation or design basis of the Chemical and Volume Control System or any other system that is described in the SAR. This modification does not add any new failure modes or operating characteristics. The failure mode described in the SAR is for the Chemical and Volume Control System 2A pump to fail to deliver the working fluid at the prevailing Reactor Coolant System pressure. The failure would reduce the redundancy of providing charging and seal water flow to the Reactor Coolant System. If this were to occur, the alternate 2B Centrifugal Charging Pump would provide the minimum flow delivery requirements at prevailing high Reactor Coolant System pressure. There are no new failure modes identified as a result of this modification. Small pipe breaks, such as tubing rupture, have been previously evaluated in the SAR. Margins associated with the integrity of the fission products barriers are not exceeded by this activity. There are no changes to any setpoint, design limit, or operating parameter. The new seals will not cause the 2A Centrifugal Charging Pump to operate outside its design basis and pose no change in the method of normal operation and emergency operation.

The scope of this modification does not involve an unreviewed safety question. No Technical Specification changes are required. No UFSAR changes are required.

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

Title: Minor Modification CE-61628, Replace the existing 2B Chemical and Volume Control System Centrifugal Charging Pump seals with new third generation seals and add a pressure gauge to monitor and trend the pump balance drum pressure

Description: This modification replaces the existing second generation Chemical and Volume Control System 2B Centrifugal Charging Pump mechanical seals with third generation seals supplied by the pump OEM, Ingersoll Dresser Pumps (IDP). The current inboard and outboard seals used on the pumps do not provide adequate service life, involve a significant risk of unrepairable shaft damage during maintenance, and are not the configurations on which active design improvements are made. IDP has shown this seal upgrade and conversion as suitable and appropriate for use with the Chemical and Volume Control System Emergency Core Cooling System (ECCS) pumps.

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

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

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

The Chemical and Volume Control System Centrifugal Charging Pump seal replacement does not change the operation or design basis of the Chemical and Volume Control System or any other system that is described in the SAR. This modification does not add any new failure modes or operating characteristics. The failure mode described in the SAR is for the Chemical and Volume Control System 2B pump to fail to deliver the working fluid at the prevailing Reactor Coolant System pressure. The failure would reduce the redundancy of providing charging and seal water flow to the Reactor Coolant System. If this were to occur, the alternate 2A Centrifugal Charging Pump would provide the minimum flow delivery requirements at prevailing high Reactor Coolant System pressure. There are no new failure modes identified as a result of this modification. Small pipe breaks, such as tubing rupture, have been previously evaluated in the SAR. Margins associated with the integrity of the fission products barriers are not exceeded by this activity. There are no changes to any setpoint, design limit, or operating parameter. The new seals will not cause the 2B Centrifugal Charging Pump to operate outside its design basis and pose no change in the method of normal operation and emergency operation.

The scope of this modification does not involve an unreviewed safety question. No Technical Specification changes are required. No UFSAR changes are required.

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

Title: Minor Modification CE-61630 Provide penetrations in the Equipment Hatch Seal Ring to allow access to the annulus

Description: Minor Modification CE-61630 adds two nuclear safety related penetrations in the Unit 2 Concrete Containment Equipment Hatch penetration seal ring sleeve to allow access to the annulus. This modification will involve providing two penetrations that are normally sealed by a bolted cover and gasket. The purpose of this modification is to provide another access to the annulus for routing temporary services (such as Steam Generator Maintenance, Ice Condenser Maintenance, Temporary Power, Integrated Leak Rate Testing) hoses and cables during outages. Currently, the only access to the Annulus from outside the Reactor Building during outages is through the Equipment Hatch Boot Seal. The boot seal causes a restriction on the size of hoses that can be used which is less than three inches. The Integrated Leak Rate Test requires the capacity of multiple diesel air compressors to pressurize containment during this test. Providing the new penetrations will allow larger hoses to be used for the Integrated Leak Rate Test which will result in quicker pressurization of containment. In addition, providing the new penetrations will help prevent future degradation of the boot seal. This degradation occurs when the boot seal is unbolted and moved to allow hoses and cables to be routed to the annulus.

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

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

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

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

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

Calculation CNC-1144.02-04-0002, Rev. 2 has been performed to qualify the design of the two new penetrations. During normal operation, the penetration openings will be sealed by a nuclear safety related cover plate and gasket that is bolted to the Concrete Containment Equipment Hatch penetration seal ring sleeve. A divider bar will remain in the center of the opening to satisfy the security requirement to maintain the area of the penetrations less than 96 square inches. The cover plate, bolting, divider bar and gasket materials are suitable for the design conditions in the annulus and equipment hatch penetration. The gasket material is the same material used for the boot seal.

Evaluation: Based on calculation CNC-1144.02-04-0002, Rev. 2, the function or integrity of the Concrete Containment Equipment Hatch penetration seal ring sleeve will not be affected by the new penetrations C401 and C402. The penetration design (cover plate, bolting, divider bar and gasket) has been qualified for the applicable design loads (seismic, annulus ventilation system pressure, tornado pressure). The design or function of any other reactor building structure will not be affected.

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

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

Title: Minor Modification CE-61681, Relocate valve 2SM-44

Description: Valve 2SM-44 is a 1-1/2" Anchor Darling Globe valve with a Limitorque motor operator. The valve is installed in a vertical run of pipe. This valve has a history of leaking by the seat and leaking externally which results in loss of unit efficiency. The valve was disassembled and inpected during the 2EOC10 Refueling Outage. The inspection showed that the valve body had extensive damage in the seating area which required replacement of the valve body and stem/disc assembly. The packing leak was caused by scoring on the stem. The damaged parts indicate that the damage occurred due to the orientation of the valve. The valve instruction manual recommends installation of this valve with the valve stem vertical. The design of the T-Head connection between the disc and stem allows enough movement in the disc to drop (due to gravity since the valve stem is horizontal) and catch on an edge internal to the valve body. This valve will be relocated to a horizontal run of pipe which is just downstream of valve 2SM-044.

The primary purpose of the Main Steam System is to convey steam from the steam generators to the high pressure turbine. In addition, the Main Steam System supplies steam as required for: the main and auxiliary feedwater pump turbines, the condenser steam air ejectors, the main and feedwater pump turbine seals, miscellaneous auxiliary equipment, and the second stage of the moisture separator reheaters.

The main turbine stop valves (2SM-54, 2SM-55, 2SM-56, and 2SM-57) are operated by hydraulic actuators and close on turbine trip to stop main steam flow into the turbine. A steam drain is located at each stop valve inlet to prevent condensate from entering the turbine. The above seat area of each main turbine stop valve is provided with a warmup drain to the condenser. Each drain features an electric motor operated globe valve which is opened during startup and shutdown. These valves are 2SM-41, 2SM-44, 2SM-43, and 2SM-42 for stop valves 2SM-54, 2SM-55, 2SM-56, and 2SM-57, respectively.

Evaluation: This modification affects valve 2SM-44, the drain valve for main steam stop valve 2SM-55 . This valve is being relocated to a horizontal run of piping to help prevent further damage to the valve and to comply with the valve manufacturers orientation recommendations. At this new location the valve will be more reliable and the possiblity of seat leakage will be reduced. The affected portion of the Main Steam System is not nuclear safety related. This modification will have no effect on the function or design basis of the Main Steam System or any other system. Valve 2SM-44 will continue to perform its design function. The design requirements of this portion of the Main Steam System are the same at the existing location and at the new location. There are no electrical changes associated with the valve relocation. There is no seismic or stress analysis associated with this portion of the system. The affected piping is supported in accordance with the construction specification CNS-1206.00-04.0002. There is no weight change associated with the valve relocation but the existing construction supports will be evaluated to ensure that they are still adequate. This modification does not involve an unreviewed safety question. No changes to the Technical Specifications are required. No UFSAR changes are required.

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

Title: Minor Modification CE-61683, Replace Nuclear Service Water System piping to the Spent Fuel Pool Cooling System with more corrosion resistant piping material.

Description: Minor Modification CE-61683 will replace some Nuclear Service Water System piping with material more resistant to corrosion. Microbial Induced Corrosion (MIC) is evident in the four inch stainless steel piping that provides the assured Nuclear Service Water System makeup to the Spent Fuel Cooling System. There are five indications of MIC attack in the heat-affected zones of the stainless steel welds in five separate locations along the flush connection piping between valves 2RN-159 and 2KF-097. Replacing this piping with stainless steel piping would only lead to future pinholes due to MIC. Carbon steel is not recommended because it is susceptible to general corrosion, MIC pitting and occlusion. The preferred replacement material would be superaustenitic stainless steel which has the greatest resistance to MIC. This material is the current industry choice for greatest resistance to raw water corrosion.

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

The Catawba Spent Fuel Pool Cooling System, in conjunction with the Component Cooling Water System and Nuclear Service Water System, is designed to remove heat from the Spent Fuel Pool and maintain purity and optic clarity of the pool water during fuel handling operations. The purification loop provides an alternate means for removing impurities from either the refueling cavity/transfer canal water during refueling or the Refueling Water Storage Tank water following refueling.

This modification affects the Nuclear Service Water System Assured Fuel Pool Makeup Header. The safety-related function of the Spent Fuel Pool Cooling System is to remove decay heat from the spent fuel assemblies stored in the Spent Fuel Pool and ensure that the assemblies remain covered. No active functions are required of the Spent Fuel Pool Cooling System Pumps and Heat Exchangers in the performance of this function. The inherent capacity of the water covering the fuel assemblies and nuclear safety-related Spent Fuel Pool/Liner supplemented by manual assured makeup from the safety-related Nuclear Service Water System is adequate to accomplish this safety function. This Minor Modification will have no effect on the function or design basis of the Nuclear Service Water System, the Spent Fuel Pool Cooling System or any other system. The Nuclear Service Water System and Spent Fuel Pool Cooling System will continue to perform their safety related functions. In fact, replacing the affected piping with superaustenitic stainless steel will enhance the reliability of the system by making it more resistant to raw water corrosion.

The design requirements of this portion of the Nuclear Service Water System are ASME

Section III, Class II, stainless steel, 150 psig and 150 degrees F. The new superaustenitic stainless steel will meet the design temperature and pressure requirements of the Nuclear Service Water System as stated above. Superaustenitic stainless steel is an acceptable material for use in this portion of the Nuclear Service Water System. The stress analysis for the affected portion of the Nuclear Service Water System has been evaluated to account for the piping material replacement and the existing seismic and support/restraint design is adequate.

This modification does not involve any unreviewed safety questions. No Technical Specification changes are required. No UFSAR changes are required.

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

Title: Minor Modification CE-61694 Nuclear Service Water System Cooling Control Valve change for Auxiliary Building Supply Units and Fuel Pool Supply Units

Description: Minor Modification CE-61694 will implement the following changes:

1. Place the Nuclear Service Water System cooling water control valves 1RN366 (FPSU-1), 1RN356 (ABSU-1A), 1RN361 (ABSU-1B), 2RN356 (ABSU-2A), and 2RN361 (ABSU-2B) in the fail open position continuously. During the spring, summer and fall Nuclear Service Water System cooling water flow will be provided at 175 gallons per minute (full flow) to ABSU-1A, ABSU-1B, ABSU-21A, and ABSU-2B.

When FPSU-1 is in the Nuclear Service Water System alignment in the spring, summer or fall, Nuclear Service Water System cooling water flow will be provided at 140 gallons per minute (full flow) to FPSU-1

ABSU = Auxiliary Building Supply Unit, FPSU = Fuel Pool Supply Unit

2. Remove controls for Nuclear Service Water System control valves identified in (1) above. Temperature transmitters 1(2)RNTT5930, 1(2)RNTT6000, 1(2)RNTT8690, 1(2)RNTT9970, and 1VFTT5590 will be removed. Receiver controllers 1(2)VA-RC-10, 1(2)VA-RC-10A and 1VF-RC-2 will also be removed. These transmitters and controllers are no longer needed to control operation of the Nuclear Service Water System control valves.

Evaluation: Nuclear Service Water System control valves 1RN366 (FPSU-1), 1RN356 (ABSU-1A), 1RN361 (ABSU-1B), 2RN356 (ABSU-2A) and 2RN361 (ABSU-2B) are currently used to control the flow of Nuclear Service Water System cooling water for the Auxiliary Building Ventilation System and the Fuel Pool Ventilation System air handling units. A temperature transmitter is located at the inlet of the Air Handling Unit to measure the entering air temperature, another temperature transmitter is located in the Nuclear Service Water System service water header to measure entering water temperature. The two are compared and if the Nuclear Service Water System supply water temperature is less than the entering air temperature the Nuclear Service Water System supply two way control valve is opened allowing Nuclear Service Water to flow through the cooling coil to cool the entering air. If the entering air temperature is less than the Nuclear Service Water System supply water temperature, the Nuclear Service Water System control valve is closed. The controller for the Nuclear Service Water System supply valve will operate to provide a 55 degree F supply air temperature.

This modification will place the Nuclear Service Water System control valve for each air handling unit identified above in the full open position at all times. Placing the control valves in the full open position will provide full Nuclear Service Water flow through the cooling coil. Full service water flow will carry off particulates within the service water. Currently when the Nuclear Service Water System control valve is in a throttled or closed position particulates settle and build up inside the cooling coil tubes. This blocks the flow of nuclear service water and increases tube corrosion. The full design Nuclear Service Water System flow through each ABSU cooling coil is 175 gpm, cooling water travels through 32 tubes in each section of the cooling water heat exchanger. The internal diameter of each tube is 0.527 inch. Using Velocity = Flow/Area, this gives us a full flow tube velocity of 8 ft/sec. This is well above the minimum velocity (3 ft/sec) recommended per good industrial practice. This also is less than the recommended maximum tube velocity of 8.5 ft/sec, by the Heat Exchanger Institute.

This modification will also remove Nuclear Service Water inlet temperature transmitters 1(2) VATT8690, 1(2) VATT 9970, 1(2) VATT 5930, 1(2) VATT 6000, and 1VFTT-5590 and receiver controllers 1(2)VA-RC-10, 1(2)VA-RC-10A and 1VF-RC-2. These transmitters and controllers are no longer needed to control the Nuclear Service Water System valves.

The following concerns were addressed:

1. Freezing of ABSU and/or FPSU cooling coil tubes during the cold winter months

During the winter months nuclear service water to ABSU and FPSU air handling units is shut off and the cooling coils are drained. Water from the Plant Heating Water System continues to supply hot water to the heating coils of each of these air handling units. This hot water preheats the cold outside air to 55 degrees F. and will also prevent the cooling coils from freezing during the winter months.

2. Concerns for equipment and personnel within the Auxiliary Building or the Fuel Pool Area caused by failing the Nuclear Service Water System Control Valves open.

During the winter months Nuclear Service Water is isolated from these cooling coils. The air handling units will continue to operate as they have in the past. During other seasons of the year full Nuclear Service Water flow will allow the temperature in the Auxiliary Building or the Fuel Pool Area to continue to change with changes in the outside air temperature and Nuclear Service Water System supply water temperature. Actual operation of these Nuclear Service Water System control valves has not been reliable, either due to transmitter, controller or control valve problems. The expected differences will be that the Auxiliary and Fuel Buildings will not get quite as hot during the hottest summer days. The modification can cause it to be slightly warmer in the Auxiliary Building during summer nights because the Nuclear Service Water System water temperature may be hotter than the outside air temperature, but the expected increase in temperature within the Auxiliary Building will be minimal.

3. Potential degradation of equipment within the Auxiliary Building or the Fuel Pool Area

Equipment within the Auxiliary Building is now designed for a maximum temperature of 120 degrees F. After implementation of this modification the Auxiliary Building and the Fuel Pool Building ventilation systems will continue to maintain temperatures in these areas well below the identified high and low temperature limits of the Environmental Qualification Program. One significant improvement that this modification will create is that with improved nuclear service water flow through the ABSU and FPSU air handling units, cooling will be more effective especially for the hot summer months. The net result will be that the Auxiliary Building will be cooler during the hot summer days. There will be no changes in the operation of these air handling units during the winter months of operation.

Implementation of this modification will improve the reliability of the Auxiliary Building Ventilation System and the Fuel Pool Ventilation System and will have no effect on any of the accidents evaluated in the UFSAR. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. No UFSAR changes are required.

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

Title: Minor Modification CE-61730, Provide Low Pressure Service Water System Thirty Inch Connection and Valve 2RL-845

Description: The Low Pressure Service Water System to Condenser Circulating Water System piping has become fouled which is causing reduced makeup capability to the Cooling Towers when Low Pressure Service Water System header pressures are in the 65 psig range. An alternate makeup line is necessary to provide additional makeup capacity for the Unit 2 Cooling Towers. This modification will install a thirty inch connection on the Low Pressure Service Water System "B" Train thirty-six inch header with a thirty inch piping connection. This connection will include thirty inch isolation valve 2RL-845 and will potentially be used to provide temporary makeup to the Unit 2 Cooling Towers per a Temporary Modification if needed before the permanent Low Pressure Service Water System to Condenser Circulating Water System Makeup Line is installed. Note that this connection should be used as part of the permanent new makeup line.

The Low Pressure Service Water System is designed to supply low pressure service water for various makeup and cooling functions on the secondary side of the plant. The Low Pressure Service Water System consists of the following: Low Pressure Service Water pumps, Low Pressure Service Water strainers, intake and discharge structures, and piping, valves, and instrumentation. The three fifty percent capacity Low Pressure Service Water pumps (one for each unit and a shared spare) located on the Low Pressure Service Water intake structure on Lake Wylie discharge to two Low Pressure Service Water strainers. Normally, one strainer will serve each unit, but if one strainer is out of service for any reason, the other strainer is capable of handling full Low Pressure Service Water flow. Downstream of the strainers, two pipes (one for each unit) supply water to the various Unit 1, Unit 2 and shared equipment such as: Cooling Towers for makeup, Main Turbine Lube Oil Coolers, Generator Stator Water Coolers, Hydrogen Coolers, Recirculated Cooling Water Coolers, Heating Water System Sample Coolers, etc. After removing rejected heat in the above coolers, the Low Pressure Service Water discharge combines with the Nuclear Service Water System discharge and is returned to Lake Wylie through two pipes (one for each unit) at the discharge structure.

The Low Pressure Service Water System does not perform any safety function and is therefore not assigned a safety class. Low level radioactive liquid wastes from various tanks in the Liquid Radwaste System will be discharged through a radiation monitor to the Nuclear Service Water System discharge piping in the Auxiliary Building. The Nuclear Service Water System discharge will combine with the Low Pressure Service Water System discharge and will be returned to Lake Wylie at the Low Pressure Service Water System discharge structure.

This Modification affects the Low Pressure Service Water System Supply Header "B" piping in the yard. This piping supplies various plant equipment as noted above. The affected Low Pressure Service Water System piping is Duke Class H (non-nuclear safety related, non-seismically designed). The connection provided by this minor modification will not affect the capability of the Low Pressure Service Water System to supply various plant equipment. This connection will potentially be used to supply additional makeup capability for the Unit 2 Cooling Towers. If this additional makeup is needed, a temporary modification will be processed to provide the design for the temporary makeup line to the Unit 2 Cooling Towers. The effect the temporary makeup line has on the Low Pressure Service Water System will be evaluated per the temporary modification process. Also, the permanent new makeup line should utilize this connection and will be evaluated per the minor modification that provides the new makeup line. This Minor Modification only provides the connection to the existing Low Pressure Service Water System and will have no effect on the function or design basis of the Low Pressure Service Water System or any other system.

The design requirements of this portion of the Low Pressure Service Water System are Duke Class H, carbon steel, 125 psig and 88 degrees F. The thirty inch piping is carbon steel and will meet the design temperature and pressure requirements of the Low Pressure Service Water System as stated above. Carbon steel is an acceptable material for use in this portion of the Low Pressure Service Water System. Valve 1RL-843 is stainless steel and also and will meet the design temperature and pressure requirements of the Low Pressure Service Water System. Stainless steel is an acceptable material for use in this portion of the Low Pressure Service Water System. The affected Low Pressure Service Water System piping has been evaluated to account for the additional weight of the valve and piping assembly and new supports will not be required.

Evaluation: There is no unreviewed safety question associated with this modification. The modification will have no effect on the probability or consequences of accidents analyzed in the UFSAR. No changes to the Technical Specifications are required. UFSAR Figure 9-63 (Low Pressure Service Water System Flow Diagram CN-1575-1.0) will be revised to reflect the addition of the new connection added by this modification.

83 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-61741, Clear alarm condition on 1MC6 1AD29-1.3 by disconnecting signal conductors and changing annunciator logic

Description: Modification CE-61741 will clear an alarm condition on control room annunciator 1AD29-1.3 (Liquid Waste Evaporator Panel Trouble) located on main control board 1MC6 by disconnecting the annunciator signal conductors and changing the annunciator logic to require a close contact for an alarm status. Also annunciator 1AD29-1.3 will be spared.

Evaluation: Seven Radwaste alarm panels are monitored by Control Room personnel through annunciator 1AD29 located on control board 1MC6. Since the "ENABLE/DEFEAT" switch for the annunciator is presently in the "DEFEAT" position, these alarms are not monitored by Control Room personnel. Chemistry is working on resolving the current alarm indications so that when the annunciator is enabled, all alarm conditions will be cleared. The Waste Evaporator Skid that is monitored by 1AD29-1.3 is "out-of-service", therefore, resulting in a continuous alarm condition when 1AD29 is returned to the "ENABLE" condition. This minor modification will clear this alarm condition. Therefore, eliminating a nuisance alarm in the Control Room when 1AD29 is in "ENABLE". This annunciator does not serve a safety function. No new failure modes are created as a result of this minor modification. UFSAR Chapter 11, Section 11.2.2.6 "System Instrumentation and Control" will be revised. No changes to the Technical Specifications are required. A 10CFR50.59 evaluation concluded that this change could be made without prior NRC approval.

73 **Type:** Minor Modification **Unit:** 1

Title: Minor Modification CE-61742, Add a lifting device in Unit 1 Upper Containment for flange for Steel Containment Vessel Penetration MK M371

Description: A safety concern was identified associated with the removal and reinstallation of spare upper containment penetration flange MK. M371. This penetration is used for access of temporary power, air supply and ice condenser hoses during refueling outages. Minor Modification CE-61742 will provide a lifting device (davit assembly), attached to the penetration sleeve, on the "inside containment" side of the Unit 1 Steel Containment Vessel, to allow safe removal and replacement of the penetration blind flange.

Evaluation: Penetration MK. M371, at the Steel Containment Vessel (SCV) only, is in the cylinder wall portion of the SCV. This steel cylinder supports the Steel Containment Dome and functions as a pressure boundary for the primary containment. The SCV is designed to limit the release of energy and radioactivity in the event of a Design Basis Event. The davit assembly will be attached to the existing sleeve for this twenty inch penetration, on the "inside containment" side only. It will be used to manipulate the blind flange on the end of the sleeve when the flange is removed/installed.

The penetration and the SCV are nuclear safety related. The attachment of the davit assembly to the penetration sleeve is also nuclear safety related. The remainder of the davit assembly is not nuclear safety related, but is seismically designed.

The davit assembly for this penetration will be identical those previously installed on two similar penetrations, MK. M234 and MK. M452.

The installation of this item will have no effect on the probability or consequences of accidents analyzed in the UFSAR. The containment penetration davit arm assembly is only a lifting device used for rigging/lifting of the penetration blind flange during refueling outages when it is opened for maintenance activities. The weld of the davit assembly to the sleeve will not degrade the sleeve. An inspection will be performed on the attachment area to verify there will be no leakage through the new weld area. Also, the applied torsional moment to the sleeve when the davit is loaded will not degrade the sleeve. This modification does not affect the operation of the penetration.

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

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

Title: Minor Modification CE-61743, Add a lifting device in Unit 21 Upper Containment for flange for Steel Containment Vessel Penetration MK M371

Description: A safety concern was identified associated with the removal and reinstallation of spare upper containment penetration flange MK. M371. This penetration is used for access of temporary power, air supply and ice condenser hoses during refueling outages. Minor Modification CE-61743 will provide a lifting device (davit assembly), attached to the penetration sleeve, on the "inside containment" side of the Unit 2 Steel Containment Vessel, to allow safe removal and replacement of the penetration blind flange.

Evaluation: Penetration MK. M371, at the Steel Containment Vessel (SCV) only, is in the cylinder wall portion of the SCV. This steel cylinder supports the Steel Containment Dome and functions as a pressure boundary for the primary containment. The SCV is designed to limit the release of energy and radioactivity in the event of a Design Basis Event. The davit assembly will be attached to the existing sleeve for this twenty inch penetration, on the "inside containment" side only. It will be used to manipulate the blind flange on the end of the sleeve when the flange is removed/installed.

The penetration and the SCV are nuclear safety related. The attachment of the davit assembly to the penetration sleeve is also nuclear safety related. The remainder of the davit assembly is not nuclear safety related, but is seismically designed.

The davit assembly for this penetration will be identical those previously installed on two similar penetrations, MK. M234 and MK. M452.

The installation of this item will have no effect on the probability or consequences of accidents analyzed in the UFSAR. The containment penetration davit arm assembly is only a lifting device used for rigging/lifting of the penetration blind flange during refueling outages when it is opened for maintenance activities. The weld of the davit assembly to the sleeve will not degrade the sleeve. An inspection will be performed on the attachment area to verify there will be no leakage through the new weld area. Also, the applied torsional moment to the sleeve when the davit is loaded will not degrade the sleeve. This modification does not affect the operation of the penetration.

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

82 Type: Minor Modification **Unit:** 1

Title: Minor Modification CE-61749, Provide Drain Valve 1RL843

Description: In order to perform pipe inspections, cleanings, valve replacements, and other modifications to the Low Pressure Service Water System, the manway just upstream of valve 1RL-A7 must be opened to provide access to the the Low Pressure Service Water System "B" Train Supply Header. Since existing drains are inadequate, the Low Pressure Service Water System piping at the manway cannot be drained as needed for the removal of the manway. A new and larger drain line will be located on the ten inch piping just upstream of valve 1RL-377. Modification CE-61749 will provide a six inch drain line and valve to aid in draining the Low Pressure Service Water System.

The Low Pressure Service Water System is designed to supply low pressure service water for various makeup and cooling functions on the secondary side of the plant. The Low Pressure Service Water System consists of the following: Low Pressure Service Water pumps, Low Pressure Service Water strainers, intake and discharge structures, piping, valves, and instrumentation.

The three 50% capacity Low Pressure Service Water pumps (one for each unit and a shared spare) located on the Low Pressure Service Water intake structure on Lake Wylie discharge to two Low Pressure Service Water strainers. Normally, one strainer will serve each unit, but if one strainer is out of service for any reason, the other strainer is capable of handling full Low Pressure Service Water flow. Downstream of the strainers, two pipes (one for each unit) supply water to the various Unit 1, Unit 2 and shared equipment such as: cooling towers for makeup, main turbine lube oil coolers, generator stator water coolers, hydrogen coolers, recirculated cooling water coolers, heating water system sample coolers, etc.

After removing rejected heat in the above coolers, the Low Pressure Service Water discharge combines with the Nuclear Service Water discharge and is returned to Lake Wylie through two pipes (one for each unit) at the discharge structure.

Evaluation: The Low Pressure Service Water System does not perform any safety function. Low level radioactive liquid wastes from various tanks in the Liquid Radwaste System will be discharged through a radiation monitor to the Nuclear Service Water discharge piping in the Auxiliary Building. The Nuclear Service Water discharge will combine with the Low Pressure Service Water discharge and will be returned to Lake Wylie at the Low Pressure Service Water discharge structure.

This Modification affects the Low Pressure Service Water System piping that supplies Heating, Ventilation and Air Conditioning (HVAC) equipment on Elevation 568 of the Service Building. The affected Low Pressure Service Water System piping is not nuclear safety related. This drain line will not affect the capability of the system to supply flow to HVAC equipment. The drain line will be used for draining the Low Pressure Service Water System as needed. This modification will have no effect on the function or design basis of the Low Pressure Service Water System or any other system.

The design requirements of this portion of the Low Pressure Service Water System are, carbon steel, 125 psig and 88 degrees F. The drain line piping and valve are carbon steel and will meet the design temperature and pressure requirements of the Low Pressure Service Water System as stated above. Carbon steel is an acceptable material for use in this portion of the Low Pressure Service Water System. The drain line and valve are suitable for draining functions. The affected Low Pressure Service Water System piping will be evaluated to account for the additional weight of the valve and piping assembly and the piping will continue to be supported in accordance with CNS-1206.00-04.002 (Design Specification for Support of Class E, G and H Piping and Temporary Piping).

This modification does not involve any unreviewed safety questions. No changes to the Technical Specifications are required. UFSAR Figure 9-63 (Low Pressure Service Water System Flow Diagram CN-1575-1.0) will be revised to reflect the addition of the new drain line added by this modification.

109 Type: Minor Modification **Unit:** 1

Title: Minor Modification CE-61886, Replacement of Flow Transmitter 1NIFT5450 (currently Rosemount Model 1151 with Rosemount Model 1153)

Description: Flow Transmitter 1NIFT5450 is currently a Rosemount Model 1151. The flow transmitter is located in a harsh environment but the Rosemount Model 1151 is not qualified for a harsh environment. Although this transmitter is not classified as nuclear safety related, it is needed for post accident analysis. The transmitter will be replaced with a Rosemount Model 1153 which is qualified fir a harsh environment.

Evaluation: This modification will have no effect on any accident analyzed in the UFSAR. No plant operating procedures or test procedures are affected by this change. A 10CFR50.59 evaluation concluded that this change could be made without prior NRC approval. No Technical Specification changes are required. UFSAR Table 1-11 will be revised.

26 Type: Miscellaneous Items **Unit:** 0

Title: Catawba Nuclear Station - In Service Testing Manual, Revision 25

Description: The IST Program Manual is being revised to clarify the testing requirements for the Main Steam Isolation Valves. Currently, the IST Program requires a partial stroke at a quarterly frequency, plus full stroke testing at cold shutdown. The partial stroke will be eliminated.

This was prompted by the Improved Technical Specifications, which recommends not to partial stroke test the Main Steam Isolation Valves.

Valves 1(2)SM-1, 1(2)SM-3, 1(2)SM-5, and 1(2)SM-7 are the Main Steam Isolation Valves (MSIVs). These valves are located in the main steam piping just downstream of the Main Steam Safety Valves. The safety function of the MSIVs is to close upon initiation of a Main Steam Isolation signal to isolate the steam flow paths from the steam generators. A main steam isolation is generated by the following:

1. 2/3 steam line pressure on any steam line with 2/3 pressurizer pressure above the P-11 setpoint
2. 2/3 high-high containment pressure signal
3. 2/3 negative steam line pressure rate signals on any steam line with 2/3 pressurizer pressure below P-11 pressurizer setpoint
4. Manual

These valves are normally open, pneumatically controlled, fail closed valves.

The current testing philosophy for the MSIVs is based on the old ASME Section XI IWV Code and carried forward in the 1987 ASME ANSI OM-10 Code, which Catawba is currently using. The basic code requirement for valves in the Inservice Testing Program (IST) is to be stroke time tested quarterly. If the valves cannot be full stroke tested quarterly, then a partial stroke should be performed quarterly and a full stroke performed during cold shutdown. The code explains further, if exercising is not practicable during plant operation, it may be limited to full stroke testing at cold shutdown. Based on this, the MSIVs are partial stroke tested quarterly and full stroke tested at cold shutdown. Catawba committed to these test requirements during the initial development of the IST program in the early 1980's.

Since the ANSI/ASME Section XI OM-10 code has been in effect, the Improved Technical Specifications (ITS) were issued. Technical Specification 3.7.2 - Bases recommends against a partial stroke of these valves in modes 1 and 2, due to the potential for a spurious closure of the valve and the resulting plant transient. The Bases section goes on to say these valves should be tested in mode 3 with the unit at full temperature and pressure. The previous Technical Specification made no mention of partial stroke requirements.

UFSAR Section 10.3 discusses the Main Steam Supply System. Testing details are not discussed. Section 10.3.2 does mention the testing circuit used to perform the partial stroke, however this is merely a statement which acknowledges the existence of this test circuit.

Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing Programs" was issued in 1989 to assist utilities with development of their IST programs. In 1995, NUREG1482, "Guidelines for Inservice Testing at Nuclear Power Plants", was issued in support of Generic Letter 89-04. GL 89-04 did not address partial stroke testing. The Generic Letter was a high level document which dealt with valves on a generic basis (ie. MOVs, AOVs, check valves, etc.), and did not provide specific guidance. The NUREG on the other hand, is very specific. Section 4.2.4 of NUREG-1482, specifically discusses MSIV testing. The discussion in the NUREG reinforces the concern of introducing a plant transient by partial stroke testing MSIVs in modes 1 or 2. The NRC has endorsed this testing philosophy by the issuance of the NUREG.

Both the Licensing and Regulatory positions recognize the risk of performing a partial stroke of the MSIVs. Both also recommend only a full stroke at the appropriate time. Catawba will adopt this philosophy and full stroke test only at cold shutdown. This change to the IWV Program will be accomplished through 10CFR50.59 evaluation to cover the preceding changes. All procedure changes will be handled separately from this evaluation.

Evaluation: This change does not involve an unreviewed safety question. The change being evaluated does not change the facility as described in the UFSAR, nor does it change the test procedures, methods, or acceptance criteria of testing already being performed for these valves. Therefore the activities being evaluated (IST Program Manual change) are not tests or experiments, nor do they appear significant enough to justify inclusion in the UFSAR. No change to plant Technical Specifications is required. No UFSAR changes are required.

12 **Type:** Miscellaneous Items **Unit:** 1

Title: Erection of One Set of Suspended Scaffolding inside the Unit 1 Upper Containment Upper Ice Condenser during operation.

Description: One set of suspended scaffolding (ten feet high) will be suspended from the ice condenser hoist railings. The scaffolding will be used in support of removal/replacement of glycol coils inside the Air Handling Unit B006. The single level suspended scaffold will be constructed per the Duke Power Scaffold Manual. One aluminum walkboard (two feet wide by ten feet long) will be used as a work platform with the scaffolding assembly. This evaluation addresses the following concerns:

1) Seismic Interaction: The scaffolding will be installed and secured in accordance with the Duke Power Scaffold Manual, which will ensure seismic concerns are controlled to prevent interaction with other components/equipment in containment.

2) Post Accident Recirculation: This scaffold will be erected inside the upper ice condenser and will not be in the direct path of the six eight inch diameter penetrations in the refueling canal wall. Also, the scaffolding components will be secured such that the scaffolding cannot become loose and block these recirculation paths in case of a seismic event.

3) Aluminum in Containment: Use of aluminum walkboards in containment has been evaluated as being acceptable per engineering calculation CNC-1223.02-00-0012.

4) Combustible Material in Containment: No combustible material will be used to erect the scaffold.

5) Ice Condenser Doors (TS 3.6.13): The scaffold assembly will be suspended from the hoist railings above the door and will not interfere with free movement of the intermediate doors in the event of a Design Basis Accident (DBA).

Evaluation: Installation of the scaffolding will not increase the probability of an accident evaluated in the SAR. No new failure modes or operating characteristics are created. The scaffolding is temporary and will be removed upon completion of the work. The scaffold is temporary and its installation does not create any new credible failure modes for any systems or components in Upper Containment. The scaffolding does not change design of any other equipment in the containment. Erection of this scaffolding in the Upper Containment does not create any new safety-related equipment failure modes or operating characteristics. The scaffolding does not affect any other safety-related components or equipment. Installation of the scaffolding does not create any new credible failure modes for any containment isolation systems or associated component. The erection of a temporary scaffolding in the Upper Containment does not affect any safety limits, set points or safety system parameters.

There are no unreviewed safety questions associated with installation of this temporary scaffolding. No Technical Specification changes are required. No UFSAR changes are required.

38 **Type:** Miscellaneous Items **Unit:** 0

Title: Software and Data Quality Assurance Plan for COMET, Version 3

Description: This change involves an upgrade to software used for offline core power distribution monitoring. The current software, COMET02, is being replaced by COMET03.

Evaluation: COMET03 is an improved version of COMET02. The new software incorporates various enhancements that do not affect the fundamental methodologies used to infer power distribution or calculate Technical Specification margin.

COMET03 was certified per Duke Power's directive for software certification (NSD-800) and verified to yield the same results as COMET02, excepting the new modifications. The modifications were verified and, as applicable, are in compliance with the Technical Specifications or submitted revisions. COMET03 is, therefore, considered equivalent to COMET02.

This change involves no material changes to the plant. The COMET software and resident workstation are not part of any System, Structure, or Component (SSC) important to safety and do not directly affect any SSCs. The three systems indirectly associated, the Movable Incore Detector System, the Excore Power Range Detectors, and the Reactor Protective System, are all unaffected by this change. The only safety significant function performed by, or involved with, this system is to generate data to evaluate Technical Specifications 3.2.1, 3.2.2, and 3.2.4, and to periodically calibrate the power range AFD indications, as required per Surveillance Requirement 3.3.1.6 (Table 3.3.1-1). As indicated, the new software is functionally equivalent to the replaced software and yields the same analytical results. Therefore assurance of the fuel integrity limits associated with the referenced Technical Specifications and with the OPDT and OTDT reactor trips (AFD parameter input) are not compromised.

This change does not affect any plant parameters, safety limits or setpoints that potentially affect the fission product barriers. This change does not involve any unreviewed safety questions. No changes to the Technical Specifications or UFSAR are required.

34 **Type:** Miscellaneous Items **Unit:** 0

Title: Technical Specification 3.6.12 (Ice Bed) Bases

Description: The Catawba Technical Specification Bases associated with the ice bed portion of the ice condensers is being revised. The changes are associated with a proposed Technical Specification amendment which includes changes to two Technical Specification surveillance requirements (SRs). The first is a revision of the ice bed chemical analysis and sampling SR. The second is a revision to the ice bed flow area verification SR.

The ice bed chemical analyses and sampling change affects the current Technical Specification Bases for SR 3.6.12.3. The changes involve clarifying the methodology for the chemical analyses of the ice condenser ice bed (stored ice). Also, this Technical Specification Bases change includes clarification of a new Technical Specification SR, added by the proposed Technical Specification amendment, regarding sampling requirements for ice additions to the ice bed.

The ice bed flow area verification changes affect the current Technical Specification SR 3.6.12.2, which requires a visual inspection of the air/steam flow area within the ice condensers. This Technical Specification Bases change clarifies that the visual surveillance program is to provide a 95 percent confidence level that flow blockage does not exceed the 15 percent Technical Specification acceptance criteria. Whereas, the 0.38 inch program required inspection of as few as two flow channels per ice condenser bay, the new program will require at least 33 percent of the flow area per bay to be inspected.

The Technical Specification Bases clarifies which structures are to be inspected. The revision limits the structures to be inspected to only include "between ice baskets" and "past lattice frames and wall panels." The Technical Specification Bases revision also is expanded to explain why other structures within the ice condenser are not inspected per the SR. Also, the Westinghouse definitions for frost and ice have been added to the Technical Specification Bases to explain why frost is not an impediment to air/steam flow through the ice condenser.

Evaluation: These Technical Specification Bases changes do not involve any physical changes to the ice condenser, any physical or chemical changes to the ice contained therein, or make any changes in the operational or maintenance aspects of the ice condenser as required by the Technical Specifications. The change only adds information to the Technical Specification Bases for clarification.

There are no Unreviewed Safety Questions associated with this Technical Specification Bases change. No changes to the Technical Specifications are required. No UFSAR changes are required.

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

Title: Technical Specification Bases B 3.4.13, Reactor Coolant System Operational Leakage SR 3.4.13.1

Description: The Bases for Technical Specification 3.4.13, RCS Operational Leakage, Surveillance Requirement (SR) 3.4.13 is being revised. This SR requires the performance, every 72 hours, of a Reactor Coolant System (RCS) water inventory balance in order to verify that RCS operational leakage is within limits. The Bases for this SR are being modified to clarify that in order to provide enhanced assurance that the primary to secondary leakage limit of LCO 3.4.13 is met, a continuous calculation is performed via an Operator Aid Computer program that utilizes the ratio of primary and secondary system activities to determine a leakage rate. This verification methodology is based on guidance contained in EPRI TR-104788-R2, "PWR Primary-to-Secondary Leak Guidelines," Revision 2. In addition, on a monthly basis, verification that the primary to secondary leakage limit is not being exceeded is performed via actual sample analysis.

Evaluation: There are no Unreviewed Safety Questions associated with this change. The manner in which the plant and its accident mitigating equipment are designed, operated, and maintained was not affected by this change. No accident probabilities or consequences were affected by this change. Similarly, no probabilities or consequences of equipment malfunctions were affected. The possibility was not created for any new type of accident or equipment malfunction. No safety margins were reduced by the proposed change. No Technical Specification changes are required. No UFSAR changes are required

23 **Type:** Miscellaneous Items **Unit:** 1

Title: Temporary Modification CNTM-0059, Install Temporary Cooling for Waste Gas System Gas Analyzer Racks

Description: Temporary Modification CNTM-0059 will install a temporary source of cooling for the Waste Gas System Gas Analyzer Rack modules. The modules are used to continually sample and monitor the hydrogen/oxygen concentrations in the waste gas stream at the inlet and outlet of the Catalytic Hydrogen Recombiner skids. The Gas Analyzer Rack modules were installed per Nuclear Station Modification CN-50452 which is still in progress. Since initial installation, the analyzer probes in the Gas Analyzer Rack modules have experienced frequent failures due to high temperatures of the waste gas stream and ambient air. Temporary Modification CNTM-0059 will provide a temporary source of cooling for the modules until permanent coolers are installed as a part of Nuclear Station Modification CN-50452.

Evaluation: There are no unreviewed safety questions associated with this temporary modification. This temporary modification will have no effect on accidents analyzed in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required.

96 Type: Miscellaneous Items **Unit:** 2

Title: Temporary Modification CNTM-0063, Isolate the control power for the containment evacuation auto start signal circuitry to block nuisance alarms.

Description: Temporary Modification CNTM-0063 will isolate the control power for the containment evacuation auto-start signal circuitry to accommodate plant maintenance work on panel boards 2ERPA, 2ERPB and 2KXPA during the 2EOC11 refueling outage. This will block nuisance alarms initiated from the Radiation Monitoring System and from the Source Range Nuclear Instrumentation as a result of the panel boards down powering for corrective maintenance. It will prevent the automatic initiation of the Containment Evacuation Alarm as described in Section 7.8.4 of the UFSAR. The panel boards are being down powered in support of replacement of the of vital inverter output switches. The system will be returned to its normal configuration before unit start up.

Evaluation: This activity will have no effect on any of the accidents analyzed in Chapter 15 of the UFSAR. The ability of the radiation monitoring system to detect a high radiation condition in containment will not be degraded. Manual initiation of the containment evacuation alarm from the Control Room will still be available during this work activity. No Technical Specification changes are required. No UFSAR changes are required. A 10CFR50.59 evaluation concluded that this change could be made without prior approval form the NRC.

60 Type: Miscellaneous Items **Unit:** 0

Title: Temporary Modification CNTM-0064, Bypass Control Room Area Train A Chiller Evaporator Flow Switch

Description: Temporary modification CNTM-0064 will bypass Control Room Area Train A Chiller Evaporator Flow Switch 0YCPS6080 by installing a jumper across power supply terminals in the 1CRA-C-1 Control Panel. This flow switch provides a permissive for the chiller to start by ensuring there is a sufficient flow of chilled water through the evaporator before the compressor is allowed to start. This flow switch is causing intermittent problems and needs to be replaced. Since a qualified replacement is not currently available, this temporary modification will remain in place until a replacement is located.

Evaluation: The design basis of the Control Room Chilled Water System and the Control Room Ventilation System is to:

1. Ensure that the Control Room will remain habitable for Operations personnel during and following all credible accident conditions.
2. Ensure that the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by the system.

These functions are accomplished by pressurizing the room slightly with respect to surrounding areas, filtering 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 90 degrees F.

The Control Room Chilled Water System Chillers have an evaporator flow switch that provides a permissive for chiller start which ensures that there is sufficient flow of chilled water through the evaporator before it is allowed to start. If chilled water flow is significantly interrupted while the compressor is operating, the evaporator flow switch will stop the compressor. This will keep from freezing the water in the evaporator and possibility rupturing a tube. This flow switch is causing intermittent problems and needs to be replaced. Since a qualified replacement is not currently available, this temporary modification will remain in place until a replacement is located.

The evaporator flow switch is one of several equipment protection features installed on the A Train Chiller. The low refrigerant temperature switch also functions to prevent evaporator tube freeze up. The low refrigerant temperature switch would also trip the compressor if the chilled water flow decreased significantly because the refrigerant suction pressure and temperature would decrease to the 33 degree F. setpoint. Since the Chiller has redundant protection features, bypassing the evaporator flow switch will not adversely affect the operation of the Chiller. Flow gauge 0YCPS5010 located in the Control Room is another indicator for monitoring Chiller A conditions.

This temporary modification will have no adverse effect on the accidents evaluated in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required.

65 **Type:** Miscellaneous Items **Unit:** 0

Title: Temporary Modification CNTM-0066, Jumper 0YCTS9209B in 2CRA-C-1

Description: Temporary Modification CNTM-0066 for the Control Room Area Chiller Train B (2CRA-C-1) will install a jumper to bypass the 2CRA-C-1 Chiller low temperature load recycle switch contacts in the chiller control circuitry.

The design basis function of the combined Control Room Ventilation/Chilled Water System is to:

1. Ensure that the control room will remain habitable for Operations personnel during and following all credible accident conditions (LOCA and/or LOCA/LOOP); and
2.Ensure that the ambient air temperature does not exceed the allowable temperature for the continuous-duty rating of the equipment and instrumentation cooled by this system.

This function is accomplished by pressurizing the control room to approximately 0.125 inch water gauge (inwg) 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.

The Control Room Chilled Water System consists of two independent 100% redundant chilled water trains. Each train consists of a chilled water pump, water chiller and chilled water distribution piping system. Each control room area chiller supplies water to a control room air handling unit, control room area air handling unit, and four switchgear room air handling units.

The chilled water load recycle switch (0YCTS9209B) is a one of several equipment protection instruments installed on the Train B Chiller (2CRA-C-1). The chilled water load recycle switch functions to prevent the chiller evaporator water tubes from freezing by automatically cycling the chiller compressor on/off during low cooling load conditions. The chilled water load recycle switch has currently malfunctioned and no replacement is available. This modification will install a jumper in the chiller control circuitry to bypass the chilled water load recycle switch..

The chilled water load recycle switch is located in the chilled water supply piping system. If the chilled water temperature reaches the 38 degree F setpoint, the switch opens and the compressor automatically stops. When the chilled water supply temperature increases to approximately 48 degrees F, the switch automatically closes and the compressor automatically restarts to reduce the chilled water temperature.

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

Installation of a jumper to bypass the low chilled water temperature load recycle switch will not adversely affect any other chiller control instrumentation. The jumper will only prevent the chiller from tripping on the load recycle switch. The jumper will not prevent the chiller from operating within its design parameters.

Installation of a jumper to bypass the low chilled water load recycle switch will not adversely impact the ability of the control room area chiller to respond to a diesel generator load sequencer signal. Therefore, the ability of the chiller to perform its design basis function will not be adversely affected.

Evaluation: There are no unreviewed safety questions associated with this temporary modification because alternate means are available for protection of the chiller. No Technical Specification changes are required. UFSAR Section 7.6.13.1 will be revised.

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

Title: Temporary Modification CNTM-0067, Bypass Control Room Area Train B Chiller Evaporator Flow Switch

Description: Temporary modification CNTM-0067 will bypass Control Room Area Train B Chiller Evaporator Flow Switch 0YCPS6090 by installing a jumper across power supply terminals in the 2CRA-C-1 Control Panel. This flow switch provides a permissive for the chiller to start by ensuring there is sufficient flow of chilled water through the evaporator before the compressor is allowed to start. This flow switch is causing intermittent problems and needs to be replaced. Since a qualified replacement is not currently available, this temporary modification will remain in place until a replacement is located.

Evaluation: The design basis of the Control Room Chilled Water system and the Control Room Ventilation System is to:

1. Ensure that the Control Room will remain habitable for Operations personnel during and following all credible accident conditions.

2. Ensure that the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by the system.

These functions are accomplished by pressurizing the room slightly with respect to surrounding areas, filtering 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 90 degrees F.

The Control Room Chilled Water Systems have an evaporator flow switch that provides a permissive for chiller start which ensures that there is sufficient flow of chilled water through the evaporator before it is allowed to start. If chilled water flow is significantly interrupted while the compressor is operating, the evaporator flow switch will stop the compressor. This will keep from freezing the water in the evaporator and possibility rupturing a tube. This flow switch is causing intermittent problems and needs to be replaced. Since a qualified replacement is not currently available, this temporary modification will remain in place until a replacement is located.

The evaporator flow switch is one of several equipment protection features installed on the B Train Chiller. The low refrigerant temperature switch also functions to prevent evaporator tube freeze up. The low refrigerant temperature switch would also trip the compressor if the chilled water flow decreased significantly because the refrigerant suction pressure and temperature would decrease to the 33 degree F. setpoint. Since the Chiller has redundant protection features, bypassing the evaporator flow switch will not adversely affect the operation of the Chiller. Flow gauge 0YCPS5010 located in the Control Room is another indicator for monitoring Chiller B conditions.

This temporary modification will have no adverse effect on accidents evaluated in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required.

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

Title: Temporary Modification CNTM-0068, Tie in to Demineralized Water System Sample Line for Chemical Addition during 2EOC11

Description: Temporary Modification CNTM-0068 will add a tee and an isolation valve to the Demineralized Water System at the outlet of valve 1YM-194 to allow addition of wet lay-up chemicals to the discharge of the deaerator which will be in discharge to the Steam Generators.

The vacuum deaerators for the Demineralized Water System are designed to supply deaerated water to the Steam Generators and to the Reactor Makeup Water Storage Tank. Normally the Steam Generators are filled during refueling outages through the Auxiliary Feedwater System with chemical additions made by the Chemical Addition System through the Steam Generator Wet Layup System. During refueling outage 2EOC11 there are anticipated times when these systems will not be available. This modification will be performed in order to minimize personnel radiation dose and to maintain a corrosion preventive environment in the steam generators.

The addition of water via the Demineralized Water System is a design function of the system which has been previously analyzed although the addition of chemicals in this manner has not been addressed . The chemicals that will be added are carbohydrazide and 3 methoxypropylamine (3-MPA). These are the same chemicals that are normally fed through the Chemical Addition System. Injecting these chemicals in the Water Treatrnent Room will allow thorough mixing prior to reaching the Steam Generators.

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

Evaluation: This temporary modification is being installed in the non-nuclear safety related Conventional Sampling System. This modification is being made to allow the introduction of chemicals into the Steam Generators during refueling outage 2EOC11 in the event the normal chemical injection point is unavailable. There are no safety related instruments or components affected by the implementation of this temporary modification. The temporary modification will be installed while the unit is at power but will be used when the Steam Generators are being filled. Based on the timing of this evolution and previously analyzed use of the chemicals, no accident probability is affected. There are no unreviewed safety questions associated with this temporary modification. No UFSAR changes are required. No Technical Specification changes are required.

51 **Type:** Miscellaneous Items **Unit:** 2

Title: Temporary Station Modification (TSM) Work Order 98376683 (01/02) , Bypass the P-12 Interlock in Mode 4 for extended cooldown on Condenser Steam Dump Valves

Description: Temporary Station Modification (TSM) Work Order 98376683 provides the method for bypass of the P-12 interlock and provides a method to use two additional (Banks 2 and 3) condenser steam dump valve banks for unit cooldown while in procedure OP/2/A/6100/002. The P-12 interlock will be bypassed in the Auxiliary Safeguards Cabinets (ASC) to disable the interlock when appropriate pressure and temperature conditions are met during Unit 2 cooldown in Mode 4. Technical Specification 3.3.2 requires that the interlock be operable during Modes 1, 2, and 3.

This interlock may be bypassed when the unit is in Mode 4 since it is no longer required by Technical Specifications. The condenser steam dumps are controlled using the Steam Pressure Controller before and after the P-12 interlock is bypassed. This controller can be operated in automatic with a steam pressure setpoint or in manual with a pushbutton demand signal. This procedure reduces the amount of time the Residual Heat Removal System is needed to operate during unit cooldown by performing an extended cooldown using condenser dump valves at lower temperatures. This method of cooldown is expected to reduce the amount of CRUD precipitated upon start of the Residual Heat Removal System and lower general area dose rates during shutdown.

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

1. The ability to add positive reactivity at a faster rate than would be possible using only three steam dump 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 the TSM is utilized (ability to cool with all three banks of steam dump valves at 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 2 given in Technical Specification Figure 3.4.3-2 will be afforded by the six additional dump valves' heat removal capability.

Both items 1 and 2 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 the TSM but the effects are different with the P-12 interlock bypassed on the other two banks of steam dump valves. All nine steam dump 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/2/A/6100/002 has been revised to include provisions for either Mode 6 boron concentration or the boron concentration associated with the final temperature of the cooldown prior to utilizing the TSM to make available all three banks of steam dump valves for cooldown. Thus, adequate shutdown margin will be maintained and return to criticality is not possible.

2. An evaluation was performed to assess the cooldown potential following failure of the steam dump controller after utilizing the TSM (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 steam dump 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 deg F) and the associated opening of just one bank (Bank #1) of steam dump valves results in a much worse cooldown by comparison. Thus, PTS events will not be exacerbated by this alternate cooldown method.

Evaluation: There are no unreviewed safety questions associated with the use of TSM Work Order 98376683. No Technical Specification changes are required. Section 10.4.4.2 of the UFSAR was revised per Minor Modification CE-61595 which installs permanent key operated switches in the Aux Safeguards Cabinets to accomplish the same functions as the TSM installation does under Work Order 98376683. This will be the last time TSM Work Order 98376683 is needed following implementation of CE-61595 during the next Unit 2 outage, U2EOC 11. No additional UFSAR changes are needed.

107 Type: Miscellaneous Items **Unit:** 1

Title: Temporary Station Modification Work Order 98417226 for CNTM-0082 Bypass P-12 Interlock in Mode 4 for Extended Cooldown on Condenser Steam Dump Valves

Description: Temporary Station Modification Work Order 98417226 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.

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 heat removal capability of the six additional dump valves 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 degrees 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 heat removal capability of six additional dump valves.

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 degrees 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.

Evaluation: A 10CFR50.59 evaluation determined that Temporary Station Modification Work Order 98417226 could be implemented without prior approval from the NRC. No Technical Specification changes are required. Section 10.4.4.2 of the UFSAR may be revised after the implementation of Minor Modification CE-61594, which will install permanent key switches in the Aux Safeguards Cabinets to accomplish the same function as this Temporary Station Modification Work Order.

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

Title: Nuclear Station Modification CN-11390/00 Variable Speed Fuel Transfer System Upgrade

Description: Modification CN-11390/00 will increase the speed of the Fuel Transfer System, reducing the cycle in each direction. The existing constant speed fuel transfer system is supplemented by upgrades to achieve a variable speed system. The existing motor and gear drive assembly is replaced with a variable speed motor and appropriate angle gear assembly. Original Westinghouse constant speed controls are modified to interface with a new variable speed control system. The variable speed system ramps up to 32 ft/min at the beginning of travel and maintains that speed for most of the transfer time. Near the end of travel, the system ramps down to 16 ft/min and stalls out on impacting the travel stop. The new control system components are housed in a wall-mounted panel adjacent to the fuel transfer control console. The upgraded system is PLC based and utilizes a variable frequency controller for speed control. Relays in the new cabinet are provided for various interlock and control functions.

Multiple on-line system monitoring features are built into the speed control to check for proper operation. Should the system fail one of these checks, the transfer rate automatically switches to slow speed, which is equivalent to the operating speed of the original system.

The SSCs added by this modification are of consistent quality and specification as the existing plant equipment to which they interface. No Appendix R concerns are created by the electrical scope of this modification. No increase in electrical load is being placed on the Blackout Power System. The structural and electrical aspects of the modification are consistent with the applicable standards such that no degradation is imposed.

Evaluation: The existing Fuel Transfer System is an on-off system that operates at 16 ft/min and travels until it contacts a hard stop and a torque switch turns off the motor. The new Fuel Transfer System will incorporate a variable frequency drive controller that is programmable and will allow for a variable drive speed as a function of track position. The new system will ramp up to a faster speed (32 ft/min) and ramp down to the current 16 ft/min impact speed at the end of the track. The new variable frequency drive introduces a new failure that could result in fuel travel speeds up to 241 feet per minute. The existing system contains no failure similar to those possible with a variable frequency drive, since it is an on-off system. An evaluation by the fuel vendors has determined that a collision with the hard stop at this maximum speed will not breach the cladding of any fuel pins within a travelling fuel assembly. Since no cladding failure is possible due to a malfunction, the event would not be deemed an accident since all other fuel handling accidents result in clad failure. Thus, no new failure modes are created that are not bounded by existing SAR evaluations. As long as the fuel transfer cart impacts the hard stop at 34 ft/min or less it can be loaded into the reactor vessel without restrictions. Also, no failure of the fuel transfer controls can result in cladding failure.

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

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

Title: Nuclear Station Modification CN-11409/00, Hot Leg Particulate Sample Panel.

Description: Modification CN-11409/00 will install a Hot Leg Particulate Sample (HLSP) Panel. The HLSP Panel will be located in the Auxiliary Building. The panel will be tied into the existing Reactor Coolant System Loop A/C Nuclear Sampling System sample line on Unit 1. Constant flow will be maintained in the sample line to support operation of the existing Nuclear Sampling System Automated Sample Panel as well as the Hot Leg Particulate Sample Panel. When required, the HLSP Panel will be aligned such that any particulate entrained in the sample flow is captured in an in-line filter device. Waste water generated in the panel during the sampling process will be hard piped to a Liquid Radwaste System drain to the Waste Evaporator Feed Tank (WEFT) Sump A. The drain from the panel spill tray will also be routed to the Liquid Radwaste System. A bottled argon source will be installed in the Nuclear Sampling System Lab to provide the panel with a sample drying medium. A sample line will be routed from the HLSP panel to the Nuclear Sampling System Sample Sink 1B to allow taking liquid grab samples. The panel will be provided a source of 110 volt AC power for temperature instrumentation.

Information with respect to crud source control, gained though the use of the HLSP Panel will be used to develop methods to limit or eliminate the effects of Axial Offset Anomalies. Axial Offset Anomalies are a problem that has occurred in many high temperature PEWR Reactor Cores since the early 1990's. An Axial Offset Anomaly is a significant core wide axial offset deviation from predictions. It is thought to be caused by accumulation of corrosion products (crud) on the surfaces of reactor core fuel assemblies.

The equipment used for the modification will be classified to the same standards of the equipment to which it interfaces. All components being added to the Nuclear Sampling System and Liquid Radwaste System will be non-nuclear safety related. The mounting of the HLSP and argon bottle will be seismically qualified. The mechanical components added to the Auxiliary Building Chilled Water System will be non-nuclear safety related. The electrical supply from the Normal AC Lighting System is non-nuclear safety related and is a 110 volt AC source for temperature instrumentation.

The tie-in point will be located on an existing Nuclear Sampling System sample line between valve 1NM-28 and the Reactor Coolant System Hot Leg Heat Exchangers 1A/1B. Manual isolation valves, 1NM-941 and 1NM-942, will control sample inlet and return. The sample outlet will be routed to the WEFT Sump A.

In order for a sample flow to exist, containment isolation valve 1NM-26B must be open. The control and accident mitigation response of this valve is unchanged by this modification. No degradation has been imposed on the containment isolation function of this line.

Evaluation: The Systems, Structures, and Components added by this modification are of consistent quality and specification as the existing plant equipment to which they interface. The design temperature and pressures are compatible. No Safe Shutdown or Appendix R concerns are created by the electrical scope of this modification. The mechanical and electrical aspects of the modification are consistent with the applicable standards such that no degradation is imposed. Thus, no new failure modes are created.

No Unreviewed Safety Questions are created as a result of modification CN-11409/00. Changes are required to UFSAR Section 9.3.2.2.1, UFSAR Figure 9-78 (Nuclear Sampling System flow diagram) will be revised. UFSAR Figures 9-81 (Nuclear Sampling System flow diagram) and 11-2 (Liquid Radwaste flow diagram) will also be revised. No Technical Specification changes are required.

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

Title: Nuclear Station Modification CN-11417/00, 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.

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 a 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.

Evaluation: The response of the Turbine Driven Auxiliary Feedwater Pump to those accidents evaluated in the UFSAR will not be affected by the 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 the modification. This is an interlock which prevents a Turbine Driven Auxiliary Feedwater Pump start upon the presence of a safety Injection Signal simultaneous with a Blackout.

This modification does not require prior approval from the NRC. No changes to the Technical Specifications are required. No UFSAR changes are required.

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

Title: Nuclear Station Modification CN-21351/00, Digital Rod Position System Upgrade

Description: Nuclear Station Modification CN-21351/00 will replace the single Digital Rod Position Indication (DRPI) System display with two separate and independent computer nodes and two CRT displays. The two new computer nodes will perform the data processing functions which were previously performed by the current DRPI display. Also, Operator Aid Computer software will be provided that will allow it to provide rod position information. These changes will essentially give three options of rod position information (two DRPI, one OAC) which could be used to meet Technical Specification surveillance requirements for those Technical Specifications requiring rod position determination (TS 3.1.4, 3.1.7). The power supply arrangement for the DRPI System will also be reconfigured. During a Loss of Off-Site Power (LOOP) Event, the current DRPI System will not indicate rod position due to loss of power since both data cabinets are powered from their respective units RPA/RPB. The current power source for the Unit 2 DRPI display is fed from panelboard 2RPA with 2RPB as the alternate source provided through a transfer switch. These panelboards are fed from a non-essential and non-battery backed power supply. Power will be reconfigured such that Unit 2A data cabinet and display will be powered from Unit 2 RPA and Unit 2B Data Cabinet will be powered from Unit 2 RPB. Additionally Unit 2B Data cabinet and display will have an alternate power supply via an automatic transfer switch from the Unit 1 RPB. This represents a change in power supply for the B Data Cabinet in that there is a backup power supply and it comes from the opposite Unit. Each separate DRPI system (A/B data cabinet and display) will have a separate power source and the B display and data cabinet has an automatic backup from the other Unit. This makes DRPI more reliable for Loss of Power (LOOP) scenarios involving Unit 2. The susceptability of losing all DRPI from a Unit 2 Loop has been removed. Half accuracy will be retained on Unit 2 from a Unit 2 LOOP since B Data cabinet and display will automatically swap to Unit 1 RPB. Additional redundancy is provided by the separate computer nodes which will improve reliability of DRPI from card and other component related failures. Also, as long as the data cabinets and one computer node are operating, the OAC can provide full accuarcy rod position indication. Loss of a single power supply to a data cabinet will still result in half accuracy DRPI indication as it did prior to this modification. The power for the Unit 1 DRPI System will be affected by this modification as described above. This is apparent since 1RPB can be powering Unit 2 DRPI equipment (data cabinet and display) in emergency backup mode and also be used as primary power for the Unit 1 DRPI B data cabinet simultaneously. Following implementation of this modification, the power arrangements will be similar to the Unit 1 configuration with the B data cabinet and display backup getting power from the opposite unit.

Evaluation: This modification has no effect on the probability or consequences of accidents evaluated in the UFSAR. The DRPI system is not nuclear safety related, does not perform any accident mitigation function, and is not an accident initiator. The ability to comply with Technical Specifications is not degraded and will actually be improved due to better fault-tolerance and redundancy of the system. There are no unreviewed safety questions associated with this modification.

No Technical Specification changes were required. UFSAR Section 3.1 and 7.7.1.3.2 were previously revised per the equivalent Unit 1 modification (CN-11351).

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

Title: Nuclear Station Modification CN-21390/00, Variable Speed Fuel Transfer System Upgrade

Description: Modification CN-21390/0 will increase the speed of the Fuel Transfer System, reducing the cycle in each direction. The existing constant speed fuel transfer system is supplemented by upgrades to achieve a variable speed system. The existing motor and gear drive assembly is replaced with a variable speed motor and appropriate angle gear assembly. Original Westinghouse constant speed controls are modified to interface with a new variable speed control system. The variable speed system ramps up to 32 ft/min at the beginning of travel and maintains that speed for most of the transfer time. Near the end of travel, the system ramps down to 16 ft/min and stalls out on impacting the travel stop. The new control system components are housed in a wall-mounted panel adjacent to the fuel transfer control console. The upgraded system is PLC based and utilizes a variable frequency controller for speed control. Relays in the new cabinet are provided for various interlock and control functions.

Multiple on-line system monitoring features are built into the speed control to check for proper operation. Should the system fail one of these checks, the transfer rate automatically switches to slow speed, which is equivalent to the operating speed of the original system.

The SSCs added by this modification are of consistent quality and specification as the existing plant equipment to which they interface. No Appendix R concerns are created by the electrical scope of this modification. No increase in electrical load is being placed on the Blackout Power System. The structural and electrical aspects of the modification are consistent with the applicable standards such that no degradation is imposed.

Evaluation: The existing Fuel Transfer System is an on-off system that operates at 16 ft/min and travels until it contacts a hard stop and a torque switch turns off the motor. The new Fuel Transfer System will incorporate a variable frequency drive controller that is programmable and will allow for a variable drive speed as a function of track position. The new system will ramp up to a faster speed (32 ft/min) and ramp down to the current 16 ft/min impact speed at the end of the track. The new variable frequency drive introduces a new failure that could result in fuel travel speeds up to 241 feet per minute. The existing system contains no failure similar to those possible with a variable frequency drive, since it is an on-off system. An evaluation by the fuel vendors has determined that a collision with the hard stop at this maximum speed will not breach the cladding of any fuel pins within a travelling fuel assembly. Since no cladding failure is possible due to a malfunction, the event would not be deemed an accident since all other fuel handling accidents result in clad failure. Thus, no new failure modes are created that are not bounded by existing SAR evaluations. As long as the fuel transfer cart impacts the hard stop at 34 ft/min or less it can be loaded into the reactor vessel without restrictions. Also, no failure of the fuel transfer controls can result in cladding failure.

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

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

Title: Nuclear Station Modification CN-21392/00, Abandon Positive Displacemet Pump No. 2

Description: This modification will abandon-in-place the Positive Displacement Pump (PDP) No. 2 and associated Chemical and Volume Control System, Nuclear Service Water System, and Component Cooling System piping and components in Unit 2. Wiring for power, instrumentation, and control will be deleted. Interfacing controls associated with PDP No. 2 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. 2 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 2NV-476 will be removed and the valve gagged closed. The Electric Motor Operator (EMO) for valve 2NV-477 will be electrically disconnected and the EMO will remain installed on the valve. Valve 2NV-477 will be closed and can be operated with the handwheel on the EMO. Valve 2NV-481 will be pneumatically disconnected and the limit switches removed. Valve 2NV-481 will be left closed and cannot be operated unless an air supply is connected to the valve. Valve 2NV-478 will no longer need to be locked throttled and will be shown on the flow drawing as normally closed. Relief valve 2NV-305 will be gagged closed since its purpose was overpressure protection for PDP No. 2.

Instrumentation associated with valve 2NV-477 (position control on main control board) and speed control for PDP No. 2 will be deleted from Control Board 2MC10. 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: Modification CN-21392/00 does not involve an Unreviewed Safety Question. The PDP No. 2 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 9-3, Table 9-6, Table 9-22, Table 9-23, and Table 12-19.

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

Title: Nuclear Station Modification CN-21394/00, Upper Internals Guide Tube Support (Split Pin) Replacement Project

Description: Modification CN-21394/00 will replace the upper internal guide tube support pins, which are known as split pins. Stress corrosion cracking of these split pins has occurred in the industry. These failures are not a nuclear safety issue, only an economic one. Loose parts due to failures have been discovered at other Westinghouse plants. The currently installed split pins are fabricated from Inconel X-750 Rev. A and each pin has the appearance of a "clothes pin". A split pin has two separate legs that have an interference fit with a hole in the upper core plate. The split pins are attached to the bottom flange of a guide tube assembly via its shank, nut and locking device. The pins are captured with a washer welded to the bottom of the guide tube support. The material for the replacement split pin is SA-316CW which is qualified and approved for this application. The modification to the upper internals include the following activities:

1. Replace all split pins (a quantity of 146) with cold worked type 316 stainless steel material.
2. Remove all guide tube/flow column hold down bolts (a quantity of 536) and install approximately 50% (4 vs. 8) of the hold down bolts fabricated from cold worked type 316 stainless steel material, except for the guide tubes adjacent to the outlet nozzles. The guide tubes located adjacent to the outlet nozzles will have all 8 replacement bolts installed.
3. Remove the 15 x 15 guide tube flexures (a quantity of 24) and removable inserts (a quantity of 6) and replace them with mechanical plugs.
4. Remove the small orifice cover plates (a quantity of 4) from 17 x 17 plutonium recycle guide tubes
5. Remove the large orifice cover plates (a quantity of 8) from the 17 x 17 guide tubes and replace with like material.

Note: Items 2 through 5 have to be removed in order to access the split pins to effect replacement. It is cheaper and easier to install new bolts (item 2) than to reuse existing bolts. The flexures (item 3) have caused problems in some plants so removal and replacement with plugs is desirable. Item 4 was possible due to item 3 plug installation. Item 5 is a remove/replace exercise with like material.

Evaluation: An evaluation was performed, WCAP-15252 Rev 1, to document the acceptability of the replacement CW316 support pin material and pin design. The engineering evaluation documents the acceptability of the pin and nut material and assembly. In so doing, WCAP-15252 Rev 1 documents the functional equivalency of the replacement support pin and nut assembly relative to the previous components. Also, the functional equivalency of the associated replacement hardware such as the guide tube hold down cap screws, guide tube cover plate hardware (which are essentially like-in-kind replacements), is provided. The reactor components being modified are nuclear safety related components. The split pin modification is a nuclear safety related modification. The replacement split pins are procured as nuclear safety related material.

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

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

Title: Nuclear Station Modification CN-21399/00, Hotwell Level Instrumentation Range Extension, New Valve 2CM-33 Control Switch and Autoclosure of Valve 2CS-47 on Auxilairy Feedwater System Autostart

Description: Modification CN-21399/00 will implement changes to the Condensate Systems that are related to the reliability of Auxiliary Feedwater System suction sources. Valve CM-33 will have a control switch added that allows the operators take manual control of the valve to OPEN/CLOSE or place it in AUTO. Also, an automatic action is being added to valve CS-47 to close on an Auxiliary Feedwater System autostart signal along with a reset button. A note will be added on the flow diagram for valves CS-33 and CS-57 that will show them as normally isolated whenever valve CM-127 is isolated to prevent a loss of Auxiliary Feedwater System inventory from the Upper Surge Tank. In addition a new hotwell level instrument will be added with an extended range.

Valve CM-33 is an automatic valve that opens and aligns a flowpath from the hotwell to the Upper Surge Tank on high discharge pressure from the hotwell pumps or high level in the hotwell. This valve automatically controls hotwell level during power maneuvers by opening to allow the hotwell pumps to deliver excess hotwell inventory to the Upper Surge Tank. It also provides a hotwell pump minimum flow path.

Even though this modification is providing a control switch for manual operation of valve CM-33, the ability to use this flowpath is dependent on unisolating the flowpath to CM-33 by opening valve CM-32 and its bypass valve CM-146 shown on UFSAR Figure 10-18 (Unit 1 equivalent drawing). Minor Modification CE-10831 currently relies on the path through valve CM-35 for manual make-up. A future change (either modification or procedure change) will be necessary to utilize the path through valve CM-33 by opening valve CM-32 and its bypass valve CM-146.

Manual valves CS-32 and CS-35 are contained in the emergency make-up line from the Upper Surge Tank to the hotwell and valve CS-32 isolates automatic valves CS-33 and CS-57. A flow diagram note will be added that shows valves CS-32 and CS-35 as normally closed whenever valve CM-127 is isolated to prevent a loss of Auxiliary Feedwater System inventory from the Upper Surge Tank. The emergency make-up lines serve as a recirculation path back to the hotwell when valve CM-127 is routing flow to the Upper Surge Tank. By closing valves CS-32 and CS-35, the emergency make-up line is effectively isolated. This action provides further assurance that the Upper Surge Tank, as a suction source for the Auxiliary Feedwater System, is preserved and protected from certain failures (non-safety failures of Air Operated Valves CS-33 and CS-57 closure) resulting in Upper Surge Tank depletion. Normal Hotwell make-up on lo-level is still available from the Upper Surge Tank via valve CS-47 discussed above. Thus, the unit is still able to tolerate condensate transients that reduce Hotwell level within the capability of the CS-47 flowpath. Any challenges beyond this are deemed to result in a Auxiliary Feedwater System actuation anyway (e.g. Feedwater Line Break). This modification isolates valves as shown on flow diagram CN-2590-2.0 rev 9A which is the Unit 2 equivalent of UFSAR Figure 9-59 (Unit 1 flow diagram).

The new control switch for valve CM-33 and reset switches, one for each train of

Auxiliary Feedwater System autostart, for valve CS-47 have been qualified for addition to the Control Board. No Appendix R concerns are created. The Safe Shutdown capability of the plant is not degraded by this modification. The changes made by this modification will improve the reliability of the Auxiliary Feedwater System by isolation of valves CS-33/ CS-57 and autoclosure of valve CS47 on Auxiliary Feedwater System autostart. Both of these changes preserve Upper Surge Tank inventory, which is preferred as a suction source over the hotwell, when needed.

Evaluation: No unreviewed safety questions are created by modification CN-21399/0. No Technical Specification changes are required. UFSAR text changes were made to section 10.4.1.5 and section 10.4.7.5.2 to describe the isolation of valves CS-33 and CS-57 under the analogous Unit 1 modification, CN-11399/00. UFSAR Figure 9-59 (Condensate Storage System flow diagram) was also revised accordingly to describe the isolation status of valves CS-33 and CS-57. Also, changes to section 10.4.7.5.2 were made to describe the manual control switch for valve CM-33. No additional UFSAR changes are required as a result of this modification.

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

Title: Nuclear Station Modification CN-21400, Additional Temperature Monitoring in the Upper Surge Tank and Hotwell and New Control Switches to Block Auxiliary Feedwater Pump Low Suction Pressure Trip

Description: Modification CN-21400/00 will implement changes to the Condensate Systems that are related to the reliability of Auxiliary Feedwater System suction sources. This modification will install new temperature instrumentation at the top of each Upper Surge Tank and provide more accurate indication of temperature near the point where recirculation from the Condensate Booster Pump discharge (Valve CM-127) is added. Also, new temperature instrumentation will be added in the condenser hotwell. Additionally, control switches will be added in the Control Room that will allow the Auxiliary Feedwater Pumps low suction pressure trip to be blocked.

Currently, the blocking of the Auxiliary Feedwater Pump low suction pressure trip is performed in the field by sliding links. The new Control Room switches will provide a more convenient and less intrusive method of blocking the trip function for better utilization of the hotwell that does not require field work or entering cabinets.

This modification involves three new Control Board switches (CA-5, CA-6, and CA-11) for which the board is qualified . There will be a switch for A Train Motor Driven Auxiliary Feedwater Pump, B Train Motor Driven Auxiliary Feedwater Pump, and the Turbine Driven Auxiliary Feedwater Pump.

UFSAR Section 10.4.9.2 "Auxiliary Feedwater System" and Section 7.4. 1 "Auxiliary Feedwater System Instrumentation and Control" were reviewed. The low suction pressure pump trip function when in manual control is not described in the text. It appears on UFSAR Figures 7-6 and 7-7 as a "stop pump" function and "do not align with Nuclear Service Water System" function if certain conditions exist which include manual Auxiliary Feedwater System operation and no Auxiliary Feedwater System auto-start signal.

Currently, emergency procedure, AM/2/A/5100/001, provides for defeating the low suction pressure Auxiliary Feedwater Pump trip by sliding the appropriate links. Thus, the defeat switch is not providing any capability that is not currently available. This action is only taken after resetting the Auxiliary Feedwater System following an Auxiliary Feedwater System autostart. When manual control is taken, the Nuclear Service Water System auto swap circuitry is disabled unless a valid Auxiliary Feedwater System autostart signal is generated. Also, the low suction pressure pump trip is enabled unless defeated through some means (sliding links per procedure or via switches post mod). UFSAR Figures 7-6 and 7-7 illustrate the requirements for alignment to the Nuclear Service Water Pond.

The emergency state of the plant is deemed to have been terminated when manual control is taken. Any subsequent plant conditions resulting in another Auxiliary Feedwater System autostart, such as blackout or lo-lo Steam Generator levels, would enable the Nuclear Service Water System autoswap circuitry and if the pressure setpoints are satisfied, a swap to the Nuclear Service Water System would occur. Thus, the defeat of the pump trip does not impair subsequently swapping to the Nuclear Service Water System, if conditions indicate the need, which maintains this Engineered Safety Feature. This modification neither blocks an ESF (Auxiliary Feedwater System Autostart) nor prevents an ESF (autoswap to Nuclear Service Water System) from being initiated if needed.

The additional temperature monitoring is an improvement with respect to Auxiliary Feedwater System reliability in that better utilization of Condensate sources for Auxiliary Feedwater System will be provided. The Auxiliary Feedwater temperature requirement of 138 degrees F. identified in UFSAR 10.4.9.2 can be monitored more effectively with the new temperature monitoring arrangement.

This modification installs new temperature instrumentation in the Upper Surge Tank approximately one foot from the top of the tank. This should eliminate the requirement for the Upper Surge Tank to be overflowing to the Condensate Storage Tank to prevent temperature errors since the hot water that may be of concern for Auxiliary Feedwater System operability would come into the top of the tank first. Additionally, new temperature instrumentation will be installed in the condenser hotwell approximately one foot up from the bottom of the hotwell. The accuracy of the new instrumentation will be sufficient to allow for a 136 degree F. setpoint to assure water no greater than 138 degree F. is present . The Control Room will utilize an annunciator to alarm undesirable temperature conditions in the Upper Surge Tank, Condensate Storage Tank, or hotwell.

Evaluation: No power supplies are degraded by any load changes involved with this modification. Protective devices (fuses) are adequately sized . No Appendix R concerns are created. The Safe Shutdown capability of the plant is not degraded by this modification . The temperature instrumentation will be installed with quality consistent with the systems, structures, and components to which they interface. The new Control Room switches will provide a more convenient and less intrusive method of blocking the trip function for better utilization of the hotwell. No degradation is imposed on the Auxiliary Feedwater System, or the Condensate Systems. The response of the Auxiliary Feedwater System to any Chapter 15 accident analysis for which the system is intended to respond, is unchanged by this modification. The limiting accident analysis assumes the Condensate Storage System is not available since it is non-safety related and credits proper operation of the Nuclear Service Water System for non-SBO events.

There are no Unreviewed Safety Questions associated with this modification. No Technical Specification changes are necessary. UFSAR Table 10-18 and Figure 9-59 were previously revised per the equivalent Unit 1 modification.

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

Title: Nuclear Station Modification CN-21408/00, Remove RCCA Mast

Description: Modification CN-21408/00 will remove the rod control cluster assembly (RCCA) mast from the Unit 2 Reactor Building Manipulator Crane. The RCCA mast has never been used since Catawba does not perform in-core control rod shuffling. Both Catawba units offload the entire core to the spent fuel pool during refueling outages and shuffle the control rods in the spent fuel pool. Left in place, the RCCA mast could interfere with the Reactor Vessel Temporary Cover being used during implementation of modification CN-21394, replacement of the upper internal guide tube support pins (split pins). The mast also causes visual interference when viewing fuel assemblies during core reload and when the manipulator crane is maneuvering around the reactor upper internals during reactor head work.

The mast will be mechanically removed from the manipulator crane and a cover will be installed over the hole left on the crane bridge.

The mast will be electrically disconnected from the crane and associated interlocks will be defeated or removed as necessary. All of the mast controls and indication provided on the control console are electrically disconnected and removed from the console surface. The cut-outs remaining in the console will be filled.

Evaluation: Neither the crane nor the RCCA mast performs an accident mitigation function. The modified Reactor Building Manipulator Crane is still designed to seismic requirements. The modified crane is to operate as before implementation of this modification, but without any of the interlocks related to the RCCA mast. No new requirements are being placed on the crane. A post modification retest will be performed to verify fuel handling interlocks are as expected.

UFSAR Section 15.7.4 evaluates fuel handling accidents in the containment and fuel building. The postulated fuel handling accident inside containment is the drop of a fuel assembly into the refueling cavity by the refueling machine. All the rods in the assembly are assumed to rupture. No discussion is given to the RCCA mast; therefore, there is no discussion of dropping only an RCCA (to potentially damage fuel) while using the RCCA mast. Having the RCCA mast removed does not affect the rest of the crane's ability to manipulate fuel assemblies within the UFSAR 9.1.4 design bases. The interlocks between the mast and the rest of the equipment on the crane will be removed. Post-modification testing will confirm the removal of the interlocks.

Modification CN-21408/00 does not involve an unreviewed safety question. No Technical Specification changes are identified. UFSAR Section 9.1.4.2.3 is being editorially revised to remove a superfluous parenthetical "Unit 2" note of unknown origin. UFSAR Section 9.1.4.3.1 is being revised as a result of this modification to remove a reference to the Unit 2 RCCA mast. UFSAR Figure 9-14 was previously updated per modification CN-11322 to reflect the removal of the Unit 1 RCCA Mast.

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

Title: Nuclear Station Modification CN-50452/00, Replace Oxygen and Hydrogen Analyzers with Orbisphere Analyzers

Description: Nuclear Station Modification CN-50452/00 will modify the Waste Gas System. The Hydrogen Recombiner subsystem of the Waste Gas System contains the Gas Analyzer Racks and instruments to measure Oxygen and Hydrogen concentration in the gas flow stream. This modification will replace the Gas Analyzer Racks and eliminate some instrumentation. The replacement Gas Analyzer Racks will utilize dual purpose instruments that measure both hydrogen and oxygen. The new Gas Analyzer Racks will be seismically designed although they will not be nuclear safety related. The design basis of the Hydrogen Recombiner subsystem and specifically the Gas Analyzer Racks does not include the ability to actively perform any safety functions to mitigate any Design Basis Accidents.

The new design results in a complete seismic pressure boundary with normally closed valves downstream of the drain tanks. Except when the valves are open purging the Gas Analyzer Racks to the Waste Evaporator Feed Tank Sump, no gas release could occur as a result of a seismic event as no piping failures should occur. This is a better design than the existing design which has some non-seismically designed piping normally aligned to the current nuclear safety related Gas Analyzer Racks.

Evaluation: This modification does not introduce any Unreviewed Safety Questions. No Technical Specification changes are required. UFSAR changes are required for UFSAR Figures 11-25 and 11-27, UFSAR Section 11.3, and UFSAR Table 3-4. The changes have already been included in Revision 7 (Effective Date 10/24/98) of the Catawba UFSAR.

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

Title: Nuclear Station Modification CN-50464, Nuclear Service Water System Piping Replacement to the Control Room Area Chilled Water System Chillers

Description: Nuclear Station Modification CN-50464 will change the route and piping material of the Nuclear Service Water System piping to and from the A Train and B Train Control Room Area Chilled Water System Chillers. Flow and differential pressure instrumentation will be provided for the Nuclear Service Water System flow to both chillers, with local indication only. The piping material which will be used (eight inch Schedule 40 AL6XN) has been shown to exhibit superior resistance to microbiologically induced corrosion (MIC). MIC occurs in the heat affected areas of welds and can cause piping leaks. The new AL6XN material is intended to prevent this problem. The exact piping and materials replacement, valve configuration and system boundaries are shown on several piping flow diagrams.

The valves involved in the scope of the modification will be stainless steel. Some of the eight inch butterfly valves already replaced by previous modifications, will be re-used for this modification. The differential pressure switches 0RNPS6000 and 0RNPS6040 (which provide permissives for the chillers to start) will not be used in the new configuration. They are being abandoned per minor modification CE-61172 since there is another trip that protects the chillers if Nuclear Service Water System flow is lost. A UFSAR change is required for this modification, even though the justification for removal of the trip was provided in CNCE-61172. Therefore, the associated pressure taps will not be installed in the new piping.

The Nuclear Service Water System to Component Cooling System Surge Tank assured make-up lines will have their system boundaries changed. This will be shown on the affected system flow diagrams. The Nuclear Service Water System to Nuclear Solid Waste Disposal System flush connection will be deleted by this modification since it is no longer used.

Evaluation: All affected piping has been analyzed for stress analysis and support revisions for new piping. Confirmation of abandoned piping was performed from a seismic perspective. All applicable design criteria continue to be met following this modification. A Safe Shutdown Review was performed. The ability to achieve Hot Standby and to proceed to Cold Shutdown has not been adversely affected by this modification. There are no safe shutdown or Appendix R concerns created by the electrical portion of the modification. The added equipment is of equivalent quality and specification as the equipment to which it is connected. No degradation is imposed on any system, structure or component or power supply. No adverse seismic interactions are created. No equipment important to safety is degraded by the modification. The design temperature and pressures are compatible. Per the Mechnaical Design Input Calculation, all existing failure effects requirements for the Train A and B Nuclear Service Water System to Component Cooling System are designed to perform their safety related heat removal functions in support of (1) design basis events which require ECCS operation and (2) Loss of Offsite Power.

There are no unreviewed safety questions associated with this modification. A UFSAR Change is required for Section 7.6.13.1 as a result of changes made per modification CE-

61172. UFSAR Figures 9-27 and 9-31 (Nuclear Service Water System) will be revised. UFSAR Figure 9-36 (Component Cooling System) will be revised. UFSAR Figures 9-116 and 9-117 (Control Room Area Chilled Water System) will be revised. UFSAR Figure 11-33 (Nuclear Solid Waste Disposal System) will be revised. There are no Technical Specification changes associated with this modification.

80 Type: Procedure **Unit:** 1

Title: Procedure IP/1/B/3560/005 "Electrical Heat Tracing System - Main Steam Supply System to Auxiliary Systems - Controllers" Revision 14

Description: Procedure IP/1/B/3560/005 Revision 14 will reduce the setpoint for a section of heat tracing from 450 degrees F to 425 degrees F. The heat tracing section is on piping that supplies steam to the Auxiliary Feedwater Pump Turbine #1. This section of piping is immediately upstream of the Pump Turbine. The intent of this setpoint change is to reduce the outboard bearing temperature on the Pump Turbine which has recently increased as a result of steam leakage past valves 1SA2 and 1SA5.

Evaluation: Neither the Auxiliary Feedwater Pump Turbine #1 nor the Auxiliary Feedwater System is an accident initiator as discussed in Chapter 15 of the UFSAR. Section 15.1.5 of the UFSAR discusses steam system piping failures. However, the procedure changes addressed by this evaluation will not result in any increase in the frequency of occurrence of the failure of any steam system piping. Decreasing the heat tracing temperature actually reduces the likelihood of a piping failure.

The Auxiliary Feedwater Pump Turbine #1 is part of the Auxiliary Feedwater System which is designed to remove decay heat. The system is designed to allow a cooldown of the plant from normal operating temperatures and pressures to the point that the Residual Heat Removal System may be placed in service thus assuring long term decay heat removal. The Auxiliary Feedwater Pump Turbine #1 also functions as part of the Standby Shutdown System. In this role it 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 cooldown from the Control Room or the Auxiliary Shutdown Panels. The turbine receives steam from electrically heat traced piping originating from the B and C Steam Generators. The Auxiliary Feedwater Pump Turbine #1 is a nuclear safety-related component, which receives Engineered Safeguards System auto-start signals on loss of offsite and all emergency AC power, and steam generator low-low level on 2 of 4 steam generators. The Auxiliary Feedwater Pump Turbine #1 is required to mitigate several UFSAR Chapter 15 accident scenarios.

The primary concern associated with decreasing the heat trace setpoint temperature is maintaining the operability of Auxiliary Feedwater Pump Turbine #1. Reducing the heat trace setpoint temperature will result in increased condensate formation in the piping. Accumulation of large volumes of condensate in the piping could potentially cause water hammers, turbine damage or turbine overspeed trips. However, the amount of additional condensate formed as a result of the heat trace setpoint change will be managed by the available surge volume in the drain piping, as well as the blowdown capacity of the two available orifices (1SAOR0031 and 1SAOR0038). Revision 14 to IP/1/B/3560/005 will not result in an increase in the likelihood of occurrence of a malfunction of the Auxiliary Feedwater Pump Turbine #1 or any other system, structure, or component important to nuclear safety.

Procedure OP/1/A/6250/002 revision 114 presents one concern related to ensuring the operability of the Auxiliary Feedwater Pump Turbine #1. Opening valve 1SA11 will allow additional condensation formed in the piping to blow down through orifice

1SAOR0038 to the Turbine Exhaust System drains. Condensate could accumulate in the piping drains to the Liquid Radwaste System Auxiliary Feedwater Pump Turbine #1 Sump, which is located in the Auxiliary Feedwater Pump Turbine #1 pit. The concern is that increased hot condensate drainage to the sump will result in elevated pit ambient temperatures. Operators may be required to enter the pit, during, or shortly after a Auxiliary Feedwater Pump Turbine #1 run. Elevated ambient temperatures in the pit could prevent personnel access and, ultimately result in Auxiliary Feedwater Pump Turbine #1 inoperability.

It has been determined that the rate that condensate is admitted to the sump will not appreciably increase as a result of opening valve 1SA11. The horizontal twelve inch turbine exhaust drain pipe located downstream of 1SA11 is open to atmosphere via the turbine exhaust. The twelve inch pipe acts as a turbine drain collection header, which drains though the Liquid Radwaste System Sump. This configuration provides that back pressure in the turbine drain system piping will essentially be unaffected by the pressure in the 1SA11 drain line. The maximum expected back pressure in the turbine drain system piping, due to the turbine running, is approximately 3 psig. The volume of the turbine drain system pipe is much greater than the expected increase in condensate formation. Based on the configuration of the drain piping, it will always drain at approximately the same rate when the turbine is running. In addition, any subsequent steam flow through 1SAOR0038 will mix with the steam that has passed though the turbine. The amount of steam that may pass on to the sump will not increase. Opening valve 1SA11 will have a negligible effect on the pit temperature.

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

25 **Type:** Procedure **Unit:** 0

Title: Procedure OP/1(2)/A/6200/006, Revision 38 (Unit 1), Revision 42 (Unit 2) addressing Opening Primary Pressure Boundary Check Valve Bypass Lines

Description: Procedure OP/1(2)/A/6200/006, Revision 38 (Unit 1), Revision 42 (Unit 2), adds an enclosure to address reseating secondary Safety Injection System /Reactor Coolant System pressure boundary check valves by pressurizing between the primary and secondary check valves using Reactor Coolant System pressure. Opening the primary check valve bypass isolation valve establishes the maximum amount of differential pressure across the ten inch cold leg accumulator (CLA) check valve, the six inch Residual Heat Removal System secondary pressure boundary valve (PBV), and the two inch Safety Injection System secondary PBV's. Establishing the maximum differential pressure across the secondary PBV's establishes a greater seating force resulting in decreased check valve leakage.

Evaluation: Primary bypass line isolation valves 1(2)NI-391, 1(2)NI-392, 1(2)NI-393, and 1(2)NI-394 are 3/4 inch air-operated, globe valves controlled from the Safety Injection System test panel located in the Auxiliary Building. The valves are normally closed with control power removed during plant Modes 1-4. Each valve fails closed on loss of air or loss of control power to the solenoid. In addition, the valve fails closed when plant operation is transferred to the Safe Shutdown Facility (SSF). Position indication for the valves is not affected by loss of control power or SSF operation. If one of these valves fails to close when required during the performance of this procedure, the operator is instructed to remove control power by using the key switch at the Safety Injection System test panel.

It is important to distinguish between a valve that fails to close and one that fails to indicate closed. The primary bypass isolation valves depend on limit switches for open and closed indication at the Safety Injection System test panel. A number of problems with limit switch indication have been encountered during outage PBV testing. The problem seen most often is that both open and closed indication for a valve is lost when the valve is stroked. Troubleshooting this kind of problem usually shows that the valve is stroking open and closed, but one or both of the limit switches are not being actuated. The design of these valves makes it highly unlikely that one would fail to close if required. The Safety Injection System test panel valves have very little travel between full open and closed. These valves are safety related and are "air to open" and if power or air is lost, an internal spring will cause the valve to close. Should it become necessary to close any particular primary bypass valve during the performance of this procedure, the fact that the valve would be operated against no differential pressure provides additional assurance of valve closure regardless of limit switch indication.

Opening a primary bypass isolation valve will cause pressure to equalize across the first ten inch PBV off the Reactor Coolant System resulting in a differential pressure of almost 2200 psig across the secondary check valves back to the Safety Injection and Residual Heat Removal Systems. The CLA secondary check valve will have almost 1600 psid. Differential pressures of at least 1400 psid are needed to obtain essentially leak tight performance from these valves due to their wide seat width. This same alignment is performed in Mode 4 at a Reactor Coolant System pressure of approximately 1300 psig during initial PBV testing prior to startup from refueling. Piping downstream of (and including) each secondary PBV is rated to withstand full Reactor Coolant System temperature and pressure. Piping sections upstream of each secondary PBV, though rated for full Reactor Coolant System pressure back through and including their motor operated isolation valves, are protected by ASME code relief valves with somewhat lower set pressures. The relief valves are in place to relieve over pressure conditions associated with limited leakage past any of the secondary PBV's. This procedure directs the operator to immediately close any primary bypass isolation valve, should Safety Injection System/Residual Heat Removal System pressure or CLA level begin increasing in an uncontrolled manner.

The allowable leakage past pressure boundary check valves is limited by Technical Specification 3.4.14. PBV's are required to close with limited leakage to prevent diversion of reactor coolant to the Safety Injection and Residual Heat Removal Systems. Also, the limited leakage provides additional assurance of valve integrity, thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Exposing the secondary pressure boundary valves to Reactor Coolant System pressure will not result in leakage in excess of Technical Specification requirements. Each of these valves was tested during the last refueling outage and leakage was verified to be less than the allowable when the test data was corrected to full Reactor Coolant System pressure. Allowing the primary check valve to float with no differential pressure does not invalidate its last test because the valve is not moving off its seat. There is no force present which could cause the primary check valves to pass flow in the forward direction. Flow in the forward direction is the only condition other than damage to a check valve that would invalidate a previously successful PBV test. Should leakage be estimated to exceed that allowed by Technical Specification 3.4.14, the unit would have to comply with the action statement for the applicable condition, isolate the leak, and proceed to a shutdown condition.

No valves other than the primary bypass isolation valves are being manipulated by this procedure. The procedure directs the operator at the Safety Injection System test panel to close any valve that may spuriously open. Double isolation is provided by normally closed manual valves in series with each air-operated valve on all but a few lines controlled from the Safety Injection System test panel. The exceptions are the CLA drain valves and the primary bypass isolation valves described above. Spurious opening or a failure to close of any CLA drain valve would pressurize the Safety Injection System accumulator fill header to CLA pressure. The header is designed to withstand full Reactor Coolant System pressure and is equipped with redundant train related containment isolation valves that close on an ESF signal, therefore this particular failure is of no concern.

Accident operation of any system that serves an ECCS function is unaffected by the performance of this procedure. Should an accident occur while the primary bypass isolation valve is open, the operator is directed by the procedure to return the valve to the closed position. If the operator is unable to perform this task, air supply to the valve would still be lost due an accident-generated Hi-Hi Containment pressure (Sp) signal that isolates the Instrument Air supply to containment. As previously described, each valve fails closed on loss of air. In any case no ECCS flow would be directed away from the core. The bypass line connects back into ten inch CLA cold leg injection line and the delay time to the core for that portion of the flow is negligible.

There is no Unreviewed Safety Question associated with this procedure change. No Technical Specification changes are required. No UFSAR changes are required.

10 Type: Procedure **Unit:** 1

Title: Procedure OP/1/A/6200/001 "Chemical and Volume Control System"

Description: Procedure OP/1/A/6200/001 "Chemical and Volume Control System" is being revised to allow unconditional use of the Unit 1 Variable Orifice.

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

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

Subsequently, the piping was reconfigured using butt welds instead of the originally installed socket welds, which significantly reduced the vibration in the piping. Valves 1(2)NV-849 trim cages were replaced with a newer version less susceptible to producing vibration. The letdown lines were retested and the vibration levels measured. The Unit 2 data gathered supports unconditional use of 2NV-849 for any allowable flow conditions using any combination of different orifices. Thus Unit 2 was returned to a condition within the original design of the letdown line. 2NV-849 can be placed in service for as long as desired. Data gathered during the testing of the Unit 1 letdown orifices during 1 EOC11 determined that the variable orifice valve 1NV-849 was acceptable to use in any combination of flow orifices and for any allowable flow except in the 80 to 110 gpm range when operating by itself. Evaluation of the data from 1EOC11 revealed that the vibration levels measured at 110 gpm using 1NV-849 by itself were probably not valid, but a flow restriction was placed in effect until the test can be repeated and new data gathered during 1EOC12. Valve 1NV-849 was re-tested by itself at 110 gpm during 1EOC12. The Unit 1 data gathered supports unconditional use of 1NV-849 for any allowable flow conditions using any combination of different orifices. Thus Unit 1 can now also be returned to a condition within the original design of the letdown line. The travel stops for both unit's NV-849 were set up for a maximum allowable flow of 110 gpm at normal Reactor Coolant System operating temperature and pressure. Thus this evaluation will support the use of NV-849 as originally designed and described in the UFSAR.

Evaluation: A search conducted in the UFSAR using the key words of "NV849", "variable orifice", and "letdown flow". The search returned several sections of the FSAR. These sections describe the function of NV-849. NV-849 is not safety-related nor is it used to mitigate any of the accident scenarios described in the UFSAR. The flow path through NV-849 is isolated by safety-related equipment that operates on safety-related signals.

The unconditional use of 1NV-849 is not an accident initiator as described in the UFSAR. This valve was included for unconditional use in the original plant design. This minor modification is to return this valve to its original purpose. Valve 1NV-849 is not a safety-related valve. Its flowpath is isolated by safety related components in the event of an accident. Thus there is no increase in the probability of occurrence of an accident previously evaluated in the SAR.

This procedure change to OP/l/A/6200/001 (Chemical and Volume Control System) does not involve an unreviewed safety question. No Technical Specifications or UFSAR changes were identified. Valve 1NV-849 controls the amount of letdown flow present in the letdown line. It does not serve any type of safety function nor is it used to mitigate any accident scenarios. Valve 1NV-849 is not a safety-related valve. Its flowpath is isolated by safety related components in the event of an accident.

71 Type: Procedure **Unit:** 1

Title: Procedure OP/1/A/6200/006M, "Safety Injection System Drain, Fill, and Vent", Revision 12

Description: Procedure OP/1/A/6200/006M (Safety Injection System Drain, Fill, and Vent), revision 12 optimizes the fill and vent operations associated with returning the Residual Heat Removal, Safety Injection and Chemical and Volume Control Systems to service following maintenance. All of the revised procedure sections are performed in no mode. Since the fuel is in the Spent Fuel Pool, the Spent Fuel Pool Cooling , Component Cooling, and Nuclear Service Water Systems are protected since they provide the cooling to the core. Thus, the fill and vent of the Residual Heat Removal, Safety Injection and Chemical and Volume Control Systems will not affect the systems that are in service to cool the core.

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

78 Type: Procedure **Unit:** 1

Title: Procedure OP/1/A/6250/002, Revision 114 "Auxiliary Feedwater System"

Description: Procedure OP/1/A/6250/002, "Auxiliary Feedwater System" and Procedure IP/1/B/3560/005 "Electrical Heat Tracing System - Main Steam Supply System to Auxiliary Systems - Controllers" are being revised to reduce the setpoint for a section of piping heat tracing from 500 degrees F. to 450 degrees F. The heat tracing involved is located on the 24 feet of piping located immediately upstream of Auxiliary Feedwater Pump Turbine #1. The intent of the setpoint change is to reduce the Auxiliary Feedwater Pump Turbine #1 outboard bearing temperature which has increased as a result of steam leakage past valves 1SA2 and 1SA5. Revision 114 to procedure OP/1/A/6250/002 will change the checklist position of valve 1SA11 from "closed" to "open". Valve 1SA11 provides isolation for the Auxiliary Feedwater Pump Turbine #1 steam supply condensate drain bypass orifice . Opening 1SA11 will increase the condensate drainage capacity of the piping in the event that reducing the heat trace setpoint results in an increase in condensate formation.

Evaluation: Neither the Auxiliary Feedwater Pump Turbine #1 nor the Auxiliary Feedwater System is an accident initiator as discussed in Chapter 15 of the UFSAR. Section 15.1.5 of the UFSAR discusses steam system piping failures. However, the procedure changes addressed by this evaluation will not result in any increase in the frequency of occurrence of the failure of any steam system piping. Decreasing the heat tracing temperature actually reduces the likelihood of a piping failure.

The Auxiliary Feedwater Pump Turbine #1 is part of the Auxiliary Feedwater System which is designed to remove decay heat. The system is designed to allow a cooldown 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 Auxiliary Feedwater Pump Turbine #1 also functions as part of the Standby Shutdown System. In this role it 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 cooldown from the Control Room or the Auxiliary Shutdown Panels. The turbine receives steam from electrically heat traced piping originating from the B and C Steam Generators. The Auxiliary Feedwater Pump Turbine #1 is a nuclear safety-related component, which receives Engineered Safeguards System auto-start signals on loss of offsite and all emergency AC power, and steam generator low-low level on 2 of 4 steam generators. The Auxiliary Feedwater Pump Turbine #1 is required to mitigate several UFSAR Chapter 15 accident scenarios.

The primary concern associated with decreasing the heat trace setpoint temperature is maintaining the operability of Auxiliary Feedwater Pump Turbine #1. Reducing the heat trace setpoint temperature will result in increased condensate formation in the piping. Accumulation of large volumes of condensate in the piping could potentially cause water hammers, turbine damage or turbine overspeed trips. However, the amount of additional condensate formed as a result of the heat trace setpoint change will be managed by the available surge volume in the drain piping, as well as the blowdown capacity of the two available orifices (1SAOR0031 and 1SAOR0038). Revision 13 to IP/1/B/3560/005 will not result in an increase in the likelihood of occurrence of a malfunction of the Auxiliary

Feedwater Pump Turbine #1 or any other system, structure, or component important to nuclear safety.

Procedure OP/1/A/6250/002 revision 114 presents one concern related to ensuring the operability of the Auxiliary Feedwater Pump Turbine #1. Opening valve 1SA11 will allow additional condensation formed in the piping to blow down through orifice 1SAOR0038 to the Turbine Exhaust System drains. Condensate could accumulate in the piping drains to the Liquid Radwaste System Auxiliary Feedwater Pump Turbine #1 Sump, which is located in the Auxiliary Feedwater Pump Turbine #1 pit. The concern is that increased hot condensate drainage to the sump will result in elevated pit ambient temperatures. Operators may be required to enter the pit, during, or shortly after a Auxiliary Feedwater Pump Turbine #1 run. Elevated ambient temperatures in the pit could prevent personnel access and, ultimately result in Auxiliary Feedwater Pump Turbine #1 inoperability.

It has been determined that the rate that condensate is admitted to the sump will not appreciably increase as a result of opening valve 1SA11. The horizontal twelve inch turbine exhaust drain pipe located downstream of 1SA11 is open to atmosphere via the turbine exhaust. The twelve inch pipe acts as a turbine drain collection header, which drains though the Liquid Radwaste System Sump. This configuration provides that back pressure in the turbine drain system piping will essentially be unaffected by the pressure in the 1SA11 drain line. The maximum expected back pressure in the turbine drain system piping, due to the turbine running, is approximately 3 psig. The volume of the turbine drain system pipe is much greater than the expected increase in condensate formation. Based on the configuration of the drain piping, it will always drain at approximately the same rate when the turbine is running. In addition, any subsequent steam flow through 1SAOR0038 will mix with the steam that has passed though the turbine. The amount of steam that may pass on to the sump will not increase. Opening valve 1SA11 will have a negligible effect on the pit temperature.

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

84 Type: Procedure **Unit:** 2

Title: Procedure OP/2/A/6200/004, " RHR System", Revision 75

Description: Procedure OP/2/A/6200/004, "RHR System", Revision 75 will allow Component Cooling System flow to either Residual Heat Removal Heat Exchanger to be manually throttled down in Mode 6 as fuel is moved from the core to the Spent Fuel Pool (automatic control will not be available). Previously full Component Cooling System flow to the Residual Heat Removal Heat Exchanger existed as long as any fuel assembly remained in the core. This flow was controlled at approximately 5000 gallons per minute. Once Component Cooling System flow to the Residual Heat Removal Heat Exchanger is reduced, Component Cooling System flow to the Spent Fuel Pool Cooling System Heat Exchangers is increased. This procedure attempts to maximize heat removal capability based on the location of the heat load.

This procedure will allow manual control of the Residual Heat Removal Heat Exchanger Component Cooling System outlet flow control valve. This is accomplished by providing a temporary sir supply line connected to the valve positioner. The air supply line has a regulator that will allow periodic adjustment of the air pressure to the valve positioner. The temporary air line is plastic tubing or a similar material and is not nuclear safety related. The pressure regulator is also not nuclear safety related. The normal air supply to this valve positioner from the Instrument Air System is also not nuclear safety related. The air line and regulator will be located so that the potential for damage is minimized. If this air line was to fail, Component Cooling System flow to the Residual Heat Removal Heat Exchanger would increase. An evaluation shows that the Component Cooling System Pumps would not be damaged by any potential near runout condition associated with this scenario.

Evaluation: This procedure change will have no adverse effect on any accident evaluated in the UFSAR. A 10CFR50.59 evaluation concluded that this change could be made without prior NRC approval. No UFSAR changes are required. No Technical Specification changes are required.

58 **Type:** Procedure **Unit:** 2

Title: Procedure PT/0/A/4200/0083, Nuclear Service Water Pond Swap Logic Test, Revision 0

Description: Procedure PT/0/A/4200/0083, Nuclear Service Water Pond Swap Logic Test, Revision 0, was written to test the relay logic associated with detecting a loss of Nuclear Service Water Pump suction from Lake Wylie. This test checks the relay logic in all possible combinations to ensure that a Loss of Lake Wylie will cause the Nuclear Service Water Pumps to start and valves to realign to an assured source of water.

The Nuclear Service Water System, including Lake Wylie and the Standby Nuclear Service Water Pond (SNSWP), is the ultimate heat sink for various nuclear safety related heat loads during normal operation and design basis events. The Nuclear Service Water System supports emergency core heat removal operation by providing cooling to the Component Cooling System via the Component Cooling System Heat Exchangers and also to the Diesel Generators via the Diesel Generator Engine Jacket Water Cooler System Heat Exchangers. Other nuclear safety related loads include the Containment Spray Heat Exchanger and Control Room Chiller Condenser. The Nuclear Service Water System also provides assured makeup to the Component Cooling System, the Spent Fuel Pool, the Auxiliary Feedwater System and the Containment Seal Water Injection System.

The Nuclear Service Water System shall be capable of mitigating the consequences of a design basis event on one unit concurrent with a loss of offsite power on both units, assuming a single failure. The most stringent design basis event for the Nuclear Service Water System is a Loss of Coolant Accident on one unit. The following assumptions are further postulated for the Nuclear Service Water System design basis LOCA:

1. Normal Shutdown of remaining unit from normal operation
2. Prolonged Drought in hot weather (maximum supply temperature/minimum supply volume)
3. Loss of Lake Wylie

In Procedure PT/0/A/4200/0083, the Nuclear Service Water System is initially aligned to the SNSWP with both Unit 1 Nuclear Service Water Pumps running. The Unit 2 Nuclear Service Water Pumps emergency 2 out of 3 low pit level automatic start contacts are disabled by opening sliding links. Various combinations of pond swap logic are initiated by opening sliding links to de-energize pond swap relays in three separate testing sequences. The pond swap logic relays are verified to reposition correctly in each of the three tests. During the last Nuclear Service Water System pit swap logic sequence, the Nuclear Service Water Pump swap contacts are enabled and verified to start the non-operating pumps when the swap logic is initiated. The valves which receive a swapover signal are monitored by measuring voltage changes in the valve circuitry that overlap IWV testing circuitry. This method allows verification that the swap signal would have actuated the valve, without actually stroking the valve. The same method is used to test the opposite train pit swap logic.

This testing is used to satisfy the requirements of NRC Generic Letter 96-01.

Evaluation: This test procedure alters the Nuclear Service Water System in two ways. The first is to alter the system from its normal alignment (to Lake Wylie) to the safe alignment (to the SNSWP). Since the Nuclear Service Water System is placed in the safe condition, there is no concern with this alignment. The second alteration is sliding links. These actions disable the automatic actuation of components based on the loss of Lake Wylie. These actions make the Nuclear Service Water Suction Transfer - Low Pit Level inoperable (Technical Specification 3.3.2 Table 3.3.2-1 Function 10) inoperable. The test procedure directs that the associated instruments be declared inoperable and ensures compliance with the Technical Specification required actions, which is to place the affected components into their safe positions. This procedure aligns the Nuclear Service Water System to the SNSWP so the required actions associated with the Technical Specification LCO are met. The procedure also disables two of the four Nuclear Service Water Pumps starting on low-low pit level (also called "emergency low"), however the Nuclear Service Water Pump automatic start on Ss, LOOP, or manual operator action is not affected. Therefore the Nuclear Service Water Pumps remain operable and the actions of Technical Specification 3.7.8 "Nuclear Service Water" are not applicable.

There are no unreviewed safety questions associated with this procedure. The procedure will not have an adverse effect on any of the accidents analyzed in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required.

45 Type: Procedure **Unit:** 0

Title: Procedure PT/0/A/4450/004A Revision 45, Auxiliary Building Ventilation System Performance Test

Description: Procedure PT/0/A/4450/004A "Auxiliary Building Filtered Exhaust System Performance Test" is being revised to correct minor discrepancies. Procedure Enclosures that place the Auxiliary Building Ventilation System in a LOCA alignment will be revised to include the check of additional Auxiliary Building Ventilation System dampers on the opposite unit to ensure the dampers aligned as designed. Enclosure 13.3 was revised to place the Auxiliary Building Ventilation System in a LOCA alignment by placing jumpers in the Diesel Generator Load Sequencer Panel 1DGLSA-2. Similar changes were made to Enclosures 13.6, 13.9, and 13.12.

Evaluation: The Auxiliary Building Ventilation System's function in the event of a design basis accident (LOCA) is to ensure that all radioactive material released from the Emergency Core Cooling System (ECCS) Pump Rooms is filtered through a HEPA and activated charcoal absorber prior to being discharged to the atmosphere. The procedure revision does not alter the operation of the Auxiliary Building Ventilation System. These procedure enclosures simulate a safety injection signal sent to the Auxiliary Building Ventilation System and verification is made that the system realigns to the ECCS Pump Rooms. The pump rooms are verified to have a negative pressure with respect to adjacent areas while the system is at its required flow rate. The enclosures that place the Auxiliary Building Ventilation System in a LOCA alignment were revised to include a check of Auxiliary Building Ventilation System dampers on the opposite unit. For example, if Train 1A is being tested, then the Auxiliary Building Ventilation System dampers on the 1A and 2A Trains will be verified to align correctly. The procedure change only adds steps to inspect these opposite train dampers. No changes are being made to the way the dampers operate. The opposite train dampers always went to the fail safe position. The procedure change only adds a step to check the damper positions. Checking damper positions will have no effect on the probability or consequences of accidents evaluated in the UFSAR.

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

48 **Type:** Procedure **Unit:** 0

Title: Procedure PT/0/A/4450/008, Revision 30 "Control Room Area Ventilation System Performance Test"

Description: Procedure PT/0/A/4450/008, Revision 30 "Control Room Area Ventilation System Performance Test" is being revised to incorporate several changes. Procedure Steps 5.6 and 5.7 were added to new Enclosures 13.15, 13.16, 13.17, and 13.18 to document what test equipment is needed to perform these enclosures. Procedure Step 6.1 was added to Section 6 to as a caution that when performing Enclosures 13.15 and 13.16 two Pressurized Filter Train Fans will be operating and extreme caution should be used when opening control room doors. Procedure Step 8.3 was revised in Enclosures 13.15 and 13.17 to reflect proper prerequisite system conditions when performing these enclosures. Procedure Step 8.4 was revised in Enclosures 13.16 and 13.18 to reflect proper prerequisite system conditions when performing these enclosures. Procedure Steps 9.6, 9.7, 9.8, and 9.9 were added to Enclosures 13.15, 13.16, 13.17, and 13.18 to describe obtaining airflow data within these enclosures. Enclosures 13.15 (Train A Test with Two Pressurizing Filter Train Fans Operating), 13.16 (Train B Test with Two Pressurizing Filter Train Fans Operating), 13.17 (Train A Control Room Pressurization Test when in a Maintenance Alignment), and 13.18 (Train B Control Room Pressurization Test when in a Maintenance Alignment) were added to Section 13 as new alignments for control room pressure boundary testing.

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 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 procedure revision does not alter the operation of Control Room Ventilation System. Procedure Enclosures 13.15 and 13.16 are performed with both Pressurization Fans in service simulating an ESF Actuation with no failures. This alignment shall gather Control Room Ventilation System performance parameters of fan flows, pressurization flow, outside air flow, and control room pressure to all adjacent areas with both intakes open, with the Unit 1 intake closed and Unit 2 intake open, and with the Unit 1 Intake open and Unit 2 intake closed. Procedure Enclosures 13.17 and 13.18 are performed with one Control Room Ventilation System train out of service, the isolation dampers closed per procedure, and the pressure boundary for the isolated Control Room Air Handling Unit and Control Room Pressurized Filter Train open simulating the routine maintenance alignment. This alignment shall also gather the same data as described for Enclosures 13.15 and 13.16.

Procedure Enclosures 13.15 and 13.16 are performed with both Pressurization Fans in service which is not the normal configuration to pressurize the Control Room. With this alignment the Control Room pressure will increase from 1.0 inwg to approximately 2.75 inwg. All Control Room doors will be electronically locked during this test. A temporary test inclined manometer will be used for obtaining Control Room pressure as described in the procedure. This pressure increase will not prohibit the operators from performing their normal duties. The jumper used to start the tested Pressurized Filter Train (PFT) will not start the opposite train Control Room Ventilation System /Control Room Chilled Water System equipment. The increased Control Room pressure will not affect any instruments located in the Control Room. During a portion of the proposed performance testing, both PFT fans will pull outside air through a single outside air intake. In this alignment the air flow rate will increase to approximately 7,200 cfm, (conservatively assumes an 80% increase in air flow with second fan operating per previous field testing). The maximum air flow velocity will be 2,400 fpm. Air flow velocity of 2,400 is common for commercial and residential ducting designs. With the increased suction from a single outside air intake the draw down or negative pressure within a single train of outside air ducting will also increase. The normal inlet pressure for the inlet ducting is less than 2 inwg, per calculation CNC- 1211.00-00-00 10. Per fan laws, doubling the entering air flow will increase the inlet suction negative pressure four times. This gives a maximum inlet negative pressure of 8 inwg. The inlet ducting is constructed of 14 ga sheet metal. Per calculation CNC- 1211.00-00-0027 this inlet ducting will withstand a negative pressure of 3 psi or 83.1 inwg. Therefore the inlet ducting will continue to operate as designed even at the higher inlet air flow created during this performance test. During the two PFT fan operation, the operation of fans will be monitored for noise and vibration to ensure stable fan operation is maintained during the test. In addition, a review of the performance fan curve shows that during normal operation the PFT fan operates at a total fan pressure of approximately 11.0 inwg. With both PFT fans operating in parallel total air flow conservatively will increase approximately 80%. Using the fan laws gives a total pressure increase per fan of approximately 4 inwg. A review of the fan performance curve shows that with an increase in total pressure up to 15 inwg both PFT fans will continue to operate in a stable portion of the fan curve. The fan performance curve also shows that the fan motor will not be overloaded. The increase in pressure will not degrade the Control Room pressure boundary. Operations will perform a Control Room differential pressure check per OP/O/A/6450/011 following the two PFT fan operation to ensure that Control Room Ventilation System has been returned to initial condition prior to test.

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

6 **Type:** Procedure **Unit:** 0

Title: Procedure PT/0/A/4450/008E, Control Room Area Chillers Performance Test, Revision 52

Description: Procedure PT/0/A/4450/008 E, Revision 52 incorporates previously approved changes along with the following new changes:

1.Provides guidance for use of a new data acquisition computer program.
2.Decreases the stabilization period between condenser head pressure setpoint changes from ten minutes to five minutes.

The only major change to the procedure is the method of data acquisition. The new computer program for data acquisition is an SDQA level B program and has been benchmarked to ensure it is accurately calculating process variables from voltage signals from the instrumentation. The new data acquisition system takes the same data and uses the same instrumentation as the previously used dLog system. The test method is unchanged.

Engineering experience has shown that the measured parameters stabilize quickly after changing the condenser head pressure setpoint. Five minutes is more than sufficient time to allow all the parameters to stabilize.

Evaluation: This procedure is being changed to allow use of a new data acquisition program. There are no Chapter 15 events which assume the Control Room Ventilation System Chillers contribute as an accident initiator. The method of performance of the procedure is not changed. Therefore, this procedure change will not increase the probability of an accident previously evaluated in the SAR. The System is placed in service and removed from service by the Operations procedure which ensures proper system parameters are present and valve lineups in place prior to starting a chiller. The procedure contains various limits and precautions, notes, and cautions to ensure the system is operated in such a manner so that there is no increase in the probability of equipment failure. The changes to this procedure do not affect the method of performance of the procedure. Therefore, the probability of a malfunction of equipment important to safety evaluated in the SAR is not increased. Administrative controls are in place to ensure that the opposite train chiller is fully operable before starting the test. In the event of a safety injection and/or reactor trip as announced by the control room, the test would be terminated and the chiller returned to operable status. Therefore, this procedure change will not increase the consequences of an accident evaluated in the SAR. The system is an accident mitigation system designed to accommodate the consequences of a single failure without loss of capability of the system to perform its intended safety function in mitigating the consequences of a Design Basis Event. No new failures are created by this procedure change. Also, only one train of the system will be affected at a time. This procedure change does not affect the method of performance of the procedure. Therefore the consequences of malfunctions of equipment important to safety evaluated in the SAR are not increased. No new failure modes are created as a result of this procedure change. No malfunctions of a different type are created by this procedure change. There is no effect on any of the fission product barriers.

This modification does not involve an unreviewed safety question. No Technical Specification changes are required. No UFSAR changes are required.

79 Type: Procedure **Unit:** 0

Title: Procedure PT/0/A/4450/008F Revision 0, "Control Room Ductwork Unfiltered Inleakage Test"

Description: Procedure PT/0/A/4450/008F Revision 0 is associated with a test to determine Control Room unfiltered inleakage. The test will install five blanks in the Control Room Ventilation System ductwork to isolate a section of ductwork that is susceptible to unfiltered inleakage. Three of the blanks will be installed on the non-operating train of the ventilation system and will be isolated from the operating train by normal maintenance procedures. Once these blanks are installed, test equipment will be connected to the ductwork. The procedure will monitor the airflow to the Control Room and Control Room Area to ensure Technical Specification 3.7.10 limits are maintained. The Technical Specification airflow limits are 6000 cubic feet per minute (cfm) of filtered pressurization air from which 4000 cfm is outside air and 2000 cfm is recirculated. This pressurization airflow maintains greater than or equal to 0.125 inwg in the Control Room relative to adjacent areas. Once the operating train design flows are verified, damper CR-D9 will be closed and serve as the pressure boundary. Once this is performed the last two blanks will be installed. The fan will be started and a constant negative pressure will be drawn on the ductwork . The unfiltered air being pulled from the ductwork will be measured. The Control Room Ventilation System train being tested will be returned to normal operation by removing the blanks per procedure PT/0/A/4450/008 F and monitoring the airflow monitors to the Control Room and Control Room Area to ensure Technical Specification 3.7.10 airflow limits are maintained.

Evaluation: The Control Room Area Ventilation System and the 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 0.125 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 will perform a test on the non operating train of the Control Room Area Ventilation System and monitor the airflow to the Control Room and Control Room Area to ensure Technical Specification 3.7.10 limits are maintained. The Technical Specification airflow limits are 6000 cfm of filtered pressurization air from which 4000 cfm is outside air and 2000 cfm is recirculated. This pressurization airflow maintains greater than or equal 0.125 inwg in the Control Room relative to adjacent areas. This activity will draw a negative pressure of 7 inwg on the train of the Control Room Area Ventilation System ductwork being tested and will cause no adverse effect on Control Room instrumentation associated with this train.

The Control Room Ventilation System is not assumed to cause any UFSAR described accidents. All changes meet the design material and construction standards of the current system. The blanks installed in the non-operating system ductwork will not adversely impact any seismic analyses. The changes being made will have negligible effect on the system and steps are included to return the system to its as-found condition. Since the proposed activity affects the Control Room Ventilation System only and the failure of that system can not cause an accident, the proposed activity will not cause any increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

The blanks to be installed on the portion of the Control Room Ventilation System ductwork that can be isolated by normal maintenance procedures will not affect the operating train since it will be isolated and therefore will not cause any failure modes to the operating train. The blanks to be installed in the shared portion of ductwork will block some of the filtered recirculation flow. To address this concern the flow of the operating system will be rebalanced by adjusting manual volume dampers so the same amount of recirculation flow is maintained. To verify proper operation of the Control Room Ventilation System system, flow rates will be measured along with the Control Room pressure. To isolate the remaining ductwork from the operating train damper CR-D-9 will be shut. CR-D-9 will serve as the new pressure boundary and allow for the remaining blanks to be installed. With the operating train fully operable and all of the blanks isolated from the operating train, no failure modes will be introduced and therefore no increases in the likelihood of an occurrence of a malfunction will occur. The Control Room Ventilation System train being tested will be returned to normal operation by removing the blanks per PT/0/A/4450/008 F and monitoring the airflow monitors to the Control Room and Control Room Area to ensure Tech Spec 3.7.10 airflow limits and Control Room pressure are maintained.

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. This activity does not meet any of the 10CFR50.59 criteria that would require a license amendment. No Technical Specification changes are required. No UFSAR changes are required.

85 **Type:** Procedure **Unit:** 0

Title: Procedure PT/0/A/4450/008G, Revision 0, "Control Room Pressurization System Configuration for Tracer Gas In-leakage Testing".

Description: Procedure PT/0/A/4450/008G Revision 0, addresses the Control Room Pressurization System Configuration for Tracer Gas In-leakage Testing. The proposed activity is a test to determine Control Room unfiltered in-leakage and measure Technical Specification 3.7.10 air flows. To determine unfiltered in-leakage, measurements are required of the outside air makeup airflow rate, the return duct airflow rate, and the supply duct airflow rate. The constant injection method is used for determining these airflows. The test will utilize the measurement of the mass flow rate of a gas stream within a duct using a tracer gas dilution technique. The tracer gas, sulfur hexafluoride (SF6), is injected into a duct at a known mass or volumetric flow rate. Grab samples are obtained downstream of the injection point and analyzed for tracer gas concentration. The ratio of the injection flow rate and the downstream sample concentrations is the volumetric flow rate in the duct. The procedure will measure the airflow to the Control Room and Control Room Area to ensure Technical Specification 3.7.10 limits are maintained. The Technical Specification airflow limits are 5,400 - 6,600 cfm of filtered pressurization air and less than or equal to 4000 cfm of outside air. The pressurization airflow maintains greater than or equal to 0. 125 inwg in the Control Room relative to adjacent areas.

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 0.125 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.

During the test, at least one train of the Control Room Area Ventilation System will be maintained fully operable. A tracer gas test will be performed to determine the amount of unfiltered in-leakage into the Control Room. The test involves the use of sulfur hexafluoride (SF6). Sulfur hexafluoride is a commonly used tracer gas when using electron capture chromatography because of its high electron affinity that provides a measurable signal. It is also transmitted and dispersed exactly like other atmospheric gases. In addition, it is non-toxic, chemically inert, odorless and tasteless. OSHA and EPA organizations have considered Permissible Exposure Limit (PEL) or Threshold Limit Value (TLV) to be a concentration of 1,000 ppm. ASTM E74-1 recommends using a concentration that is at most one tenth of the maximum safe concentration. Therefore, tracer gas injection rates shall be limited to producing less than 100 ppm within the Control Room Area Ventilation System ductwork by the vendor's procedures.

The Control Room Area Ventilation System design basis functions are to provide Control

Room pressurization and Control Room cooling. In addition, to support the site dose analysis the Control Room Area Ventilation System must limit the amount of unfiltered in-leakage into the Control Room. Performance of this procedure will not impact the ability of the Control Room Area Ventilation System to pressurize or cool the Control Room. In-leakage is only a concern during the time that neither the permanent test caps and the tracer gas equipment are installed. During this transition time when the test ports are open, minimal in-leakage into the ductwork will occur since each cap will only be off for approximately one minute. Additionally, removing and reinstalling test caps and the tracer gas equipment is a simple process and personnel are at the test port performing the installation and removal. Personnel will be ensuring that the access ports are closed as rapidly as possible during installation and removal of test equipment as instructed in procedure PT/0/A/4450/008 G. During the time that the test equipment is installed, an in-leakage concern is not created. The test equipment is air tight and is not subjected to pressures that would challenge its integrity, and the tubing is not subject to seismic concerns. Thus, this activity will not result in more than a minimal increase in consequences of an accident previously evaluated in the UFSAR.

During the tracer gas testing it is necessary to secure the Control Room doors. In the alignment with both Pressurizing Filter Train Fans operating, the Control Room pressure will increase from 1.0 to approximately 2.75 inwg. This pressure increase will not prohibit the operators from performing their normal duties. The increased Control Room pressure will not affect any instruments located in the Control Room.

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.

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

63 **Type:** Procedure **Unit:** 1

Title: Procedure PT/1/A/4200/059, Nuclear Service Water System to Auxiliary Feedwater System Piping Flush, Revision 37

Description: The Nuclear Service Water System assured makeup to the Auxiliary Feedwater System is periodically flushed per procedure PT/1/A/4200/059. The test prerequisites conservatively require four operable Nuclear Service Water System pumps and four operable diesel generators to ensure Nuclear Service Water System operability during the test. This requirement for both train related Nuclear Service Water System pumps ensures that the additional Auxiliary Feedwater System demand from the Nuclear Service Water System to Auxiliary Feedwater System train being flushed does not degrade the other essential loads assumed in the Nuclear Service Water System one pump flow balance.

The train related motor driven Auxiliary Feedwater System pump is declared inoperable during the performance of the test, since the Nuclear Service Water System assured supply cannot supply adequate flow to Auxiliary Feedwater System with the two inch drain valve open. The unit related Auxiliary Feedwater Pump Turbine remains operable, since the opposite train assured makeup is able to supply it plus the motor driven Auxiliary Feedwater System pump.

The Nuclear Service Water System to Auxiliary Feedwater System flush connects a fire hose (with adapters) on the two inch drain valve just upstream of the Nuclear Service Water System to Auxiliary Feedwater System transfer valve, 1RN-250A or 1RN-310B, and routes it to a two and one half inch connection on valve 1RN-E45 (with adapters) on the Unit 1 Nuclear Service Water System non-essential return header. Flow is established from the drain valve on the supply header to the drain valve on the return header for fifteen minutes.

The distance between the flush connections on the supply and return headers is approximately thirty feet. The fire hose will be secured at intervals to prevent excessive movement of the hose, should it fail during the flush. A walk down of the likely route for the flush hose was performed to determine the electrical cabinets near the route. The area where the flush hose will be routed is subject to water spray, therefore any Nuclear Safety Related electrical equipment in the area would be qualified for spray.

To minimize the chance of adversely affecting any electrical equipment in the area, Operators performing the flush will wrap the mechanical joints of the fire hose to suppress leakage. Also, an Operator will be stationed at one of the two flush valves for the fifteen minute flush duration. It is expected that if a hose were to fail, the flush would be secured promptly. This is reasonable because of the short distance between the two flush valves. Additionally, the flush is located in Room 400 of the Auxiliary Building, Elevation 577 feet which is a mild radiological environment during normal and post accident operation.

This test is performed to satisfy the requirements of NRC Generic Letter 89-13.

The Nuclear Service Water system, including Lake Wylie and the Standby Nuclear Service Water Pond (SNSWP), is the ultimate heat sink for various nuclear safety related heat loads during normal operation and design basis events. The Nuclear Service Water System supports Emergency Core Heat Removal operation by providing cooling to the Component Cooling system via the Component Cooling heat exchangers and also to the Diesel Generators via the Diesel Generator Engine Jacket Water Cooler system heat exchangers. Other nuclear safety related loads include the Containment Spray heat exchanger and Control Room Chiller Condenser. The Nuclear Service Water system also provides assured makeup to Component Cooling, Spent Fuel Pool, Auxiliary Feedwater supply and the Containment Seal Water Injection system.

The Nuclear Service Water System shall be capable of mitigating the consequences of a design basis event on one unit concurrent with a loss of offsite power on both units, assuming a single failure. The most stringent design basis event for the Nuclear Service Water System is a Loss of Coolant Accident on one unit. The following assumptions are further postulated for the Nuclear Service Water System design basis LOCA:

1. Normal Shutdown of remaining unit from normal operation
2. Prolonged Drought in hot weather (maximum supply temperature/minimum supply volume)
3. Loss of Lake Wylie

Evaluation: In order to perform the test, the prerequisites conservatively require that all four Nuclear Service Water System pumps and all four diesel generators are operable. Therefore, any flow diversion to the train of the Auxiliary Feedwater System being flushed is more than compensated for by the additional Nuclear Service Water System pump that is required on that train and the Nuclear Service Water System one pump flow balance is maintained. It is very conservative to require both pumps on each train, since one Nuclear Service Water System pump could handle this flow demand in addition to the other Nuclear Service Water System flows assumed in the Nuclear Service Water System one pump flow balance. Also, the Nuclear Service Water System to Auxiliary Feedwater flush procedure duration is fifteen minutes, and during this time Operators monitor the flush and are assigned to close the Nuclear Service Water System valves on the supply and return header in the case of a Safety Injection or Auxiliary Feedwater System auto start on either unit. This is reasonable since the valves are within one hundred feet of each other in the Auxiliary Building.

Flooding of the Auxiliary Building is not a concern since Operators monitor the flush during the entire fifteen minute duration. Any leakage due to a pinhole or tear in the fire hose would lead to the Operators immediately securing the flush and correcting the condition.

The seismic integrity of the Nuclear Service Water System supply and return piping is maintained while the fire hoses and adapters are attached and full of water. The additional weight has been shown to be acceptable in the piping stress analysis. Therefore, the test does not degrade the seismic integrity of the existing Nuclear Service Water System piping.

Since the seismic integrity of the Nuclear Service Water System is unaffected by the flush, and the Nuclear Service Water System flow balance is not adversely affected by the flush, connecting the ASME Section III Class III Nuclear Service Water System supply piping to the non-ASME Seismically designed Nuclear Service Water System nonessential return header piping does not create a Nuclear Service Water System Operability concern.

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

59 **Type:** Procedure **Unit:** 0

Title: Procedure PT/2/A/4200/01N, "Reactor Coolant System Pressure Boundary Leak Rate Test", Revision 40

Description: Procedure PT/2/A/4200/01N, "Reactor Coolant System Pressure Boundary Leak Rate Test", Revision 40, is used to verify that the leakage past any Reactor Coolant System Pressure Boundary Valve (PBV) does not exceed the requirements of Technical Specification Surveillance Requirement 3.4.14.1. The valves that are tested by this procedure are important in preventing over pressurization and rupture of the Emergency Core Cooling System (ECCS) low pressure piping which could result in a Loss of Coolant Accident that bypasses containment. The valves tested are the first and second stage check valves in the Safety Injection System lines to the Reactor Coolant System Cold Legs and Hot Legs as well as the Residual Heat Removal System Suction Isolation valves off Hot Leg B and Hot Leg C.

The purpose of this change is to retype the procedure into the Catawba Nuclear Station Standard Template, to standardize the procedure per the Catawba Nuclear Station Procedures Writers Manual, and to incorporate procedure enhancements and good practices.

The following changes are being made:

1. The Valve checklist for system alignment during the PBV Test has been separated into two enclosures. (Enclosure 13.33, Valve Checklist-ECCS Operating, includes valves operated from the Control Room and Safety Injection System Test Panel and Enclosure 13.33.1, Valve Checklist - Safety Injection System Test Header, are manual valves located in the Auxiliary and Reactor Building). The procedure revision also removed double verification that is not required on valves in the Reactor Building.
2. A step to ensure power disconnect switches have been returned to "DISCON" has been added to Enclosure 13.35 "Restoration".
3. A "Prerequisite System Condition" step has been added to ensure system alignment per Procedure OP/1/A/6200/009 "Cold Leg Accumulator Operation".
4. In Section 3.2 a reference was added to Technical Specification 3.5.2, "ECCS - Operating", and Technical Specification 3.5.3, "ECCS - Shutdown", which dictates when ECCS Valves must be returned to their normal alignment. Other miscellaneous references were added.
5. Various document format changes were incorporated that did not affect the technical content of the procedure or the test method.

Evaluation: UFSAR Section 6.3.4.2 "Reliability Tests and Inspections" describes the purpose of this test but does not include any details about the method of performing it. This procedure revision does not change the test method. Much of this revision concerns administrative changes.

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

64 Type: Procedure **Unit:** 2

Title: Procedure PT/2/A/4200/059, Nuclear Service Water System to Auxiliary Feedwater System Piping Flush, Revision 38

Description: The Nuclear Service Water System assured makeup to the Auxiliary Feedwater System is periodically flushed per procedure PT/2/A/4200/059. The test prerequisites conservatively require four operable Nuclear Service Water System pumps and four operable diesel generators to ensure Nuclear Service Water System operability during the test. This requirement for both train related Nuclear Service Water System pumps ensures that the additional Auxiliary Feedwater System demand from the Nuclear Service Water System to Auxiliary Feedwater System train being flushed does not degrade the other essential loads assumed in the Nuclear Service Water System one pump flow balance.

The train related motor driven Auxiliary Feedwater System pump is declared inoperable during the performance of the test, since the Nuclear Service Water System assured supply cannot supply adequate flow to Auxiliary Feedwater System with the two inch drain valve open. The unit related Auxiliary Feedwater Pump Turbine remains operable, since the opposite train assured makeup is able to supply it plus the motor driven Auxiliary Feedwater System pump.

The Nuclear Service Water System to Auxiliary Feedwater System flush connects a fire hose (with adapters) on the two inch drain valve just upstream of the Nuclear Service Water System to Auxiliary Feedwater System transfer valve, 2RN-250A or 2RN-310B, and routes it to a two and one half inch connection on valve 1RN-E45 (with adapters) on the Unit 1 Nuclear Service Water System non-essential return header. Flow is established from the drain valve on the supply header to the drain valve on the return header for fifteen minutes.

The distance between the flush connections on the supply and return headers is approximately one hundred feet. The fire hose will be secured at intervals to prevent excessive movement of the hose, should it fail during the flush. A walk down of the likely route for the flush hose was performed to determine the electrical cabinets near the route. The area where the flush hose will be routed is subject to water spray, therefore any Nuclear Safety Related electrical equipment in the area would be qualified for spray.

To minimize the chance of adversely affecting any electrical equipment in the area, Operators performing the flush will wrap the mechanical joints of the fire hose to suppress leakage. Also, an Operator will be stationed at one of the two flush valves for the fifteen minute flush duration. It is expected that if a hose were to fail, the flush would be secured promptly. This is reasonable because of the short distance between the two flush valves. Additionally, the flush is located in Room 400 of the Auxiliary Building, Elevation 577 feet which is a mild radiological environment during normal and post accident operation.

This test is performed to satisfy the requirements of NRC Generic Letter 89-13.

The Nuclear Service Water system, including Lake Wylie and the Standby Nuclear Service Water Pond (SNSWP), is the ultimate heat sink for various nuclear safety related heat loads during normal operation and design basis events. The Nuclear Service Water System supports Emergency Core Heat Removal operation by providing cooling to the Component Cooling system via the Component Cooling heat exchangers and also to the Diesel Generators via the Diesel Generator Engine Jacket Water Cooler system heat exchangers. Other nuclear safety related loads include the Containment Spray heat exchanger and Control Room Chiller Condenser. The Nuclear Service Water system also provides assured makeup to Component Cooling, Spent Fuel Pool, Auxiliary Feedwater supply and the Containment Seal Water Injection system.

The Nuclear Service Water System shall be capable of mitigating the consequences of a design basis event on one unit concurrent with a loss of offsite power on both units, assuming a single failure. The most stringent design basis event for the Nuclear Service Water System is a Loss of Coolant Accident on one unit. The following assumptions are further postulated for the Nuclear Service Water System design basis LOCA:

1. Normal shutdown of remaining unit from normal operation
2. Prolonged drought in hot weather (maximum supply temperature/minimum supply volume)
3. Loss of Lake Wylie

Evaluation: In order to perform the test, the prerequisites conservatively require that all four Nuclear Service Water System pumps and all four diesel generators are operable. Therefore, any flow diversion to the train of the Auxiliary Feedwater System being flushed is more than compensated for by the additional Nuclear Service Water System pump that is required on that train and the Nuclear Service Water System one pump flow balance is maintained. It is very conservative to require both pumps on each train, since one Nuclear Service Water System pump could handle this flow demand in addition to the other Nuclear Service Water System flows assumed in the Nuclear Service Water System one pump flow balance. Also, the Nuclear Service Water System to Auxiliary Feedwater flush procedure duration is fifteen minutes, and during this time Operators monitor the flush and are assigned to close the Nuclear Service Water System valves on the supply and return header in the case of a Safety Injection or Auxiliary Feedwater System auto start on either unit. This is reasonable since the valves are within one hundred feet of each other in the Auxiliary Building.

Flooding of the Auxiliary Building is not a concern since Operators monitor the flush during the entire fifteen minute duration. Any leakage due to a pinhole or tear in the fire hose would lead to the Operators immediately securing the flush and correcting the condition.

The seismic integrity of the Nuclear Service Water System supply and return piping is maintained while the fire hoses and adapters are attached and full of water. The additional weight has been shown to be acceptable in the piping stress analysis. Therefore, the test does not degrade the seismic integrity of the existing Nuclear Service Water System piping.

Since the seismic integrity of the Nuclear Service Water System is unaffected by the flush, and the Nuclear Service Water System flow balance is not adversely affected by the flush, connecting the ASME Section III Class III Nuclear Service Water System supply piping to the non-ASME Seismically designed Nuclear Service Water System nonessential return header piping does not create an Nuclear Service Water System Operability concern.

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

92 Type: Procedure **Unit:** 1

Title: Procedure TN/1/A/1606/MM/01I "Implementation Procedure for Minor Modification CN-61606 Work Unit 01", Revision 0

Description: Procedure TN/1/A/1606/MM/01I "Implementation Procedure for Minor Modification CN-61606 Work Unit 01" provides instructions for system isolations, instructions for relocation of instrumentation lines, and instructions for restoration and functional testing for instrument 1NDPT5090. These activities are associated with Minor Modification CE-61606 which will relocate the instrument impulse line tap for instruments 1NDPT5080 and 1NDPT5090. 1NDPT5080 monitors Residual Heat Removal Pump 1B discharge pressure and 1NDPT5090 monitors Residual Heat Removal Pump 1A discharge pressure in order to provide control indication and Operator Aid Computer input to show pump discharge conditions downstream of each pump's discharge check valve.

The Residual Heat Removal System is required to be operable in Modes 5 and 6 and is required for core decay heat removal in Mode 4. The Residual Heat Removal System also serves as a part of the Emergency Core Cooling System during the injection and recirculation phases of design basis events. A portion of the Residual Heat Removal System flow may be diverted to separate spray headers of the Containment Spray System. During normal plant operation the Residual Heat Removal System is not in service and is aligned for operations as part of the Emergency Core Cooling System. As a part of the ECCS System, both trains of the Residual Heat Removal System are required to be operable in Modes 1, 2, and 3, and one train is required operable in Mode 4.

All activities described in Procedure TN/1/A/1606/MM/01I will be performed during Modes 1,2, and 3.

Evaluation: This activity will have no effect on the probability or consequences of accidents described in the UFSAR. A 10CFR50.59 evaluation concluded that this modification can be performed without prior NRC approval. No Technical Specification changes are required. No UFSAR changes are required.

93 **Type:** Procedure **Unit:** 1

Title: Procedure TN/1/A/1606/MM/02I "Implementation Procedure for Minor Modification CN-61606 Work Unit 02", Revision 0

Description: Procedure TN/1/A/1606/MM/02I "Implementation Procedure for Minor Modification CN-61606 Work Unit 02" provides instructions for system isolations, instructions for relocation of instrumentation lines, and instructions for restoration and functional testing for instrument 1NDPT5080. These activities are associated with Minor Modification CE-61606 which will relocate the instrument impulse line tap for instruments 1NDPT5080 and 1NDPT5090. 1NDPT5080 monitors Residual Heat Removal Pump1B discharge pressure and 1NDPT5090 monitors Residual Heat Removal Pump 1A discharge pressure in order to provide control indication and Operator Aid Computer input to show pump discharge conditions downstream of each pump's discharge check valve.

The Residual Heat Removal System is required to be operable in Modes 5 and 6 and is required for core decay heat removal in Mode 4. The Residual Heat Removal System also serves as a part of the Emergency Core Cooling System during the injection and recirculation phases of design basis events. A portion of the Residual Heat Removal System flow may be diverted to separate spray headers of the Containment Spray System. During normal plant operation the Residual Heat Removal System is not in service and is aligned for operations as part of the Emergency Core Cooling System. As a part of the ECCS System, both trains of the Residual Heat Removal System are required to be operable in Modes 1, 2, and 3, and one train is required operable in Mode 4.

All activities described in Procedure TN/1/A/1606/MM/02I will be performed during Modes 1,2, and 3.

Evaluation: This activity will have no effect on the probability or consequences of accidents described in the UFSAR. A 10CFR50.59 evaluation concluded that this activity can be done without prior NRC approval. No Technical Specification changes are required. No UFSAR changes are required.

94 **Type:** Procedure **Unit:** 2

Title: Procedure TN/2/A/1607/MM/01I "Implementation Procedure for Minor Modification CN-61607 Work Unit 01", Revision 0

Description: Procedure TN/2/A/1607/MM/01I "Implementation Procedure for Minor Modification CN-61607 Work Unit 01" provides instructions for system isolations, instructions for relocation of instrumentation lines, and instructions for restoration and functional testing for instrument 2NDPT5090. These activities are associated with Minor Modification CE-61607 which will relocate the instrument impulse line tap for instruments 2NDPT5080 and 2NDPT5090. 2NDPT5080 monitors Residual Heat Removal Pump 2B discharge pressure and 2NDPT5090 monitors Residual Heat Removal Pump 2A discharge pressure in order to provide control indication and Operator Aid Computer input to show pump discharge conditions downstream of each pump's discharge check valve.

The Residual Heat Removal System is required to be operable in Modes 5 and 6 and is required for core decay heat removal in Mode 4. The Residual Heat Removal System also serves as a part of the Emergency Core Cooling System during the injection and recirculation phases of design basis events. A portion of the Residual Heat Removal System flow may be diverted to separate spray headers of the Containment Spray System. During normal plant operation the Residual Heat Removal System is not in service and is aligned for operations as part of the Emergency Core Cooling System. As a part of the ECCS System, both trains of the Residual Heat Removal System are required to be operable in Modes 1, 2, and 3, and one train is required operable in Mode 4.

All activities described in Procedure TN/2/A/1607/MM/01I will be performed during Modes 1,2, and 3.

Evaluation: This activity will have no effect on the probability or consequences of accidents described in the UFSAR. A 10CFR50.59 evaluation concluded that this modification can be performed without prior NRC approval. No Technical Specification changes are required. No UFSAR changes are required.

95 **Type:** Procedure **Unit:** 2

Title: Procedure TN/2/A/1607/MM/02I "Implementation Procedure for Minor Modification CN-61607 Work Unit 02", Revision 0

Description: Procedure TN/2/A/1607/MM/02I "Implementation Procedure for Minor Modification CN-61607 Work Unit 02" provides instructions for system isolations, instructions for relocation of instrumentation lines, and instructions for restoration and functional testing for instrument 2NDPT5080. These activities are associated with Minor Modification CE-61607 which will relocate the instrument impulse line tap for instruments 2NDPT5080 and 2NDPT5090. 2NDPT5080 monitors Residual Heat Removal Pump 2B discharge pressure and 2NDPT5090 monitors Residual Heat Removal Pump 2A discharge pressure in order to provide control indication and Operator Aid Computer input to show pump discharge conditions downstream of each pump's discharge check valve.

The Residual Heat Removal System is required to be operable in Modes 5 and 6 and is required for core decay heat removal in Mode 4. The Residual Heat Removal System also serves as a part of the Emergency Core Cooling System during the injection and recirculation phases of design basis events. A portion of the Residual Heat Removal System flow may be diverted to separate spray headers of the Containment Spray System. During normal plant operation the Residual Heat Removal System is not in service and is aligned for operations as part of the Emergency Core Cooling System. As a part of the ECCS System, both trains of the Residual Heat Removal System are required to be operable in Modes 1, 2, and 3, and one train is required operable in Mode 4.

All activities described in Procedure TN/1/A/1607/MM/02I will be performed during Modes 1,2, and 3.

Evaluation: This activity will have no effect on the probability or consequences of accidents described in the UFSAR. A 10CFR50.59 evaluation concluded that this activity can be done without prior NRC approval. No Technical Specification changes are required. No UFSAR changes are required.

57 **Type:** Procedure **Unit:** 1

Title: Procedure TT/1/A/9300/029 Revision 0, "1A Residual Heat Removal Pump Check Valve Slam Test"

Description: Temporary Test Procedure TT/1/A/9300/029 Revision 0, "1A Residual Heat Removal Pump Check Valve Slam Test", allows pressure data to be taken on the Residual Heat Removal System during pump start. This data will be used to determine the magnitude of the system pressure transient during normal Residual Heat Removal Pump starts. This test will be performed in Mode 1 with the Residual Heat Removal Pump IWP test. The test procedure installs and removes a pressure transducer on a Residual Heat Removal pump discharge piping vent valve.

Evaluation: The Residual Heat Removal System is a nuclear safety related system. Per UFSAR Section 5.4.7.1 the system transfers heat from the Reactor Coolant System to the Component Cooling System to reduce the temperature of the Reactor Coolant System to the cold shutdown temperature. The Residual Heat Removal System must be capable of cooling the Reactor Coolant System at a controlled rate during normal plant cooldown and maintains cold shutdown temperature until the plant is started up again. The Residual Heat Removal System also provides letdown for Reactor Coolant System chemistry control and auxiliary Pressurizer spray during shutdown operations. The Residual Heat Removal System serves as part of the Emergency Core Cooling System and the Containment Spray System during a design basis accident. The Emergency Core Cooling System function and the Containment Spray System function are required in operational modes 1-4.

In Temporary Test Procedure TT/1/A/9300/029 Revision 0, the Residual Heat Removal System pump is operated in the normal recirculation alignment for IWP testing. The procedure allows for installation of a non-nuclear safety related pressure transducer to a vent line located on the Residual Heat Removal System header between the Residual Heat Removal System discharge check valve and the Residual Heat Removal System Heat Exchanger. This vent valve will be opened for a short time during the Residual Heat Removal Pump start for the IWP test. The pressure transducer and associated fittings are similar in weight and size to the pipe cap which is removed from the vent, therefore the seismic analysis of the piping is not affected by the installation. The pressure transducer and associated fittings design pressure exceeds the design pressure rating of the ASME Section III Class II Residual Heat Removal System piping. The pressure transducer is connected to a non-ASME nipple downstream of vent valve 1ND-100. The transducer and fittings form a leak tight boundary similar to the normally installed pipe cap. Since the pressure transducer and the pipe cap it replaces are of similar weight and are not nuclear safety related, the attachment of the transducer and fittings will have no adverse effects on the system operation or configuration. Therefore the only issue related to this procedure is the opening of the vent valve during pump startup and pump shutdown in order to record pressure data. The procedure specifies that the vent valve will be opened minutes prior to pump start and will be returned to the normally closed position soon after data collection following pump start. The evolution of opening and reclosing the vent valve is similar to the established practice of venting the pump suction piping in order to reduce suction pressure during pump operation per procedure PT/1/A/4200/010A, "Residual Heat Removal Pump 1A Performance Test". The Residual Heat Removal

Pump and the Residual Heat Removal System are not rendered inoperable during performance of procedure PT/1/A/4200/010A.

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

47 **Type:** Procedure **Unit:** 2

Title: Procedure TT/2/A/9300/029 Revision 0, "2A Residual Heat Removal Pump Check Valve Slam Test"

Description: Temporary Test Procedure TT/2/A/9300/029 Revision 0, "2A Residual Heat Removal Pump Check Valve Slam Test", allows pressure data to be taken on the Residual Heat Removal System during pump start. This data will be used to determine the magnitude of the system pressure transient during normal Residual Heat Removal Pump starts. This test will be performed in Mode 1 with the Residual Heat Removal Pump IWP test. The test procedure installs and removes a pressure transducer on a Residual Heat Removal pump discharge piping vent valve.

Evaluation: The Residual Heat Removal System is a nuclear safety related system. Per UFSAR Section 5.4.7.1 the system transfers heat from the Reactor Coolant System to the Component Cooling System to reduce the temperature of the Reactor Coolant System to the cold shutdown temperature. The Residual Heat Removal System must be capable of cooling the Reactor Coolant System at a controlled rate during normal plant cooldown and maintains cold shutdown temperature until the plant is started up again. The Residual Heat Removal System also provides letdown for Reactor Coolant System chemistry control and auxiliary Pressurizer spray during shutdown operations. The Residual Heat Removal System serves as part of the Emergency Core Cooling System and the Containment Spray System during a design basis accident. The Emergency Core Cooling System function and the Containment Spray System function are required in operational modes 1-4.

In Temporary Test Procedure TT/2/A/9300/029 Revision 0, the Residual Heat Removal System pump is operated in the normal recirculation alignment for IWP testing. The procedure allows for installation of a non-nuclear safety related pressure transducer to a vent line located on the Residual Heat Removal System header between the Residual Heat Removal System discharge check valve and the Residual Heat Removal System Heat Exchanger. This vent valve will be opened for a short time during the Residual Heat Removal Pump start for the IWP test and for a short time when the pump is shutdown. The pressure transducer and associated fittings are similar in weight and size to the pipe cap which is removed from the vent, therefore the seismic analysis of the piping is not affected by the installation. The pressure transducer and associated fittings design pressure exceeds the design pressure rating of the ASME Section III Class II Residual Heat Removal System piping. The pressure transducer is connected to a non-ASME nipple downstream of vent valve 2ND-122. The transducer and fittings form a leak tight boundary similar to the normally installed pipe cap. Since the pressure transducer and the pipe cap it replaces are of similar weight and are not nuclear safety related, the attachment of the transducer and fittings will have no adverse effects on the system operation or configuration. Therefore the only issue related to this procedure is the opening of the vent valve during pump startup and pump shutdown in order to record pressure data. The procedure specifies that the vent valve will be opened minutes prior to pump start and will be returned to the normally closed position soon after data collection following pump start. The evolution of opening and reclosing the vent valve is similar to the established practice of venting the pump suction piping in order to reduce suction pressure during pump operation per procedure PT/2/A/4200/010A, "Residual Heat

Removal Pump 2A Performance Test". The Residual Heat Removal Pump and the Residual Heat Removal System are not rendered inoperable during performance of procedure PT/2/A/4200/010A.

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

54 **Type:** Procedure **Unit:** 0

Title: Procedures PT/0/A/4400/008A and PT/0/A/4400/008B, Nuclear Service Water System Flow Balance Train A, B (Revision 35C, 31 C).

Description: The Nuclear Service Water System, including Lake Wylie and the Standby Nuclear Service Water Pond (SNSWP), is the ultimate heat sink for various nuclear safety related heat loads during normal operation and design basis events. The Nuclear Service Water System supports emergency core heat removal operation by providing cooling to the Component Cooling System via the Component Cooling System Heat Exchangers and also to the Diesel Generators via the Diesel Generator Engine Jacket Water Cooler System Heat Exchangers. Other nuclear safety related loads include the Containment Spray Heat Exchanger and Control Room Chiller Condenser. The Nuclear Service Water System also provides assured makeup to the Component Cooling System, the Spent Fuel Pool, the Auxiliary Feedwater System and the Containment Seal Water Injection System.

The Nuclear Service Water System shall be capable of mitigating the consequences of a design basis event on one unit concurrent with a loss of offsite power on both units, assuming a single failure. The most stringent design basis event for the Nuclear Service Water System is a Loss of Coolant Accident on one unit. The following assumptions are further postulated for the Nuclear Service Water System design basis LOCA:

1. Normal Shutdown of remaining unit from normal operation
2. Prolonged Drought in hot weather (maximum supply temperature/minimum supply volume)
3. Loss of Lake Wylie

This change adds alternate acceptance criteria to the Nuclear Service Water System flow balance procedures (PT/0/A/4400/008A, Nuclear Service Water System Flow Balance Train A (Revision 35C) and PT/0/A/4400/008B, Nuclear Service Water System Flow Balance Train B (Revision 31 C)) for the Nuclear Service Water System flow to the Nuclear Service Water Pump Motor Upper Bearing Oil Coolers. The alternate acceptance criteria is for Nuclear Service Water System flow rates between 3.9 and 4.3 GPM to the Upper Bearing Oil Cooler, which is below the current 4.4 to 8.0 GPM range in the Nuclear Service Water System flow balance procedures. The alternate acceptance criteria adds temperature restrictions on the Nuclear Service Water Pump Motor bearing to the lower flow range.

The acceptance criteria in the Nuclear Service Water System flow balance procedures for Upper Bearing Oil Cooler flow is 4.4 to 8.0 GPM, which comes from the manufacturers recommended flow of 4.0 GPM to the Upper Bearing Oil Cooler. This recommended flow was incorporated into the UFSAR and Nuclear Service Water System test acceptance criteria calculation and TAC sheets. Note that the 4 GPM Nuclear Service Water Pump Upper Bearing Oil Cooler flow rate in Tables 9-2 and 9-5 of the UFSAR is the nominal or design flow rate for the Upper Bearing Oil Cooler, and will not change. The 4.0 GPM minimum flow rate was error adjusted by adding 5% of span, or 0.4 GPM, and incorporated into the Nuclear Service Water System flow balance procedures.

There is previous history of not being able to meet the 4 GPM minimum recommended flow rate to the Upper Bearing Oil Coolers. In 1987, Catawba Engineering contacted the manufacturer (Westinghouse) to provide alternate acceptance criteria since the flow rate to the Upper Bearing Oil Coolers was only 3.5 GPM at flow balance conditions. Westinghouse provided two alternate methods to verify adequate cooling water flow to the Upper Bearing Oil Coolers. Both methods were incorporated into the Nuclear Service Water System test acceptance criteria calculation as acceptable alternate acceptance criteria, but were not incorporated into the Nuclear Service Water System flow balance procedures. The alternate acceptance criteria uses Nuclear Service Water System temperature exiting the motor cooler (Case 1) or Nuclear Service Water motor bearing temperatures (Case 2). Case 2 will be incorporated into the Nuclear Service Water System flow balance procedures, with the following additional conservative measures:

When the Nuclear Service Water System flow rate to the Upper Bearing Oil Cooler is between 3.9 and 4.3 GPM, the maximum Nuclear Service Water Pump motor upper bearing temperature will be verified less than 170 degrees F. This temperature is less than the 185 degree F limit specified and is conservative. From review of Operator Aid Computer data, Nuclear Service Water Pump Motor upper bearing temperature varies up to ten degrees F with Nuclear Service Water System supply temperature seasonal variation and varying Nuclear Service Water Pump discharge flow rates. Therefore, it is conservative to have a limit of 170 degrees F. for the Nuclear Service Water pump motor upper bearing at flow balance conditions, without concern about Nuclear Service Water inlet temperature. Additionally, the minimum 3.5 GPM flow rate has been error adjusted to 3.9 GPM.

Evaluation: There are no unreviewed safety questions associated with this procedure change. The change incorporates an alternate acceptance criteria for flow to the Nuclear Service Water Pump Motor Upper Bearing Oil Cooler. The alternate criteria were furnished by Westinghouse, the manufacturer of the motor. This change will have no effect on the accidents analyzed in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required,

86 **Type:** Procedure **Unit:** 2

Title: PT/2/A/4450/001D, Revision 20, "Containment Purge Filter Train Performance Test"

Description: Procedure PT/2/A/4450/001D "Containment Purge Filter Train Performance Test" is being revised to allow the Containment Purge Ventilation System filter units to be tested during Modes 1, 2, 3 or 4. Containment Purge Ventilation System filter unit testing during a non-outage time period will reduce outage testing duration and minimize potential delays for maintenance during outages. During 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 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 filter units and prior to the filtered exhaust backdraft damper. 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 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 will be operated during Modes 1, 2, 3 or 4. Electrical jumpers will be placed to allow these fans to operate. Sliding links within an electrical panel will be opened to prevent remaining portions of the Containment Purge Ventilation System from operating including the duct heater. 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 Procedure Enclosure 13.17. The sliding links will prevent the Containment Purge supply fans from starting, Containment Isolation Valves from opening, and Reactor Building isolation dampers from opening. In addition the motor control center breakers will be opened for Containment Purge supply fans 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.

HEPA filter bank air distribution tests and Air Aerosol Mixing Uniformity tests will be performed because of major maintenance on filter unit CPFU-2A (replacement of carbon adsorber screens). This testing is required per the guidelines of ANSI N510. Historically, CNS has used the 1980 revision of ANSI N510. However, ANSI N510-1989 guidance for the air distribution tests will be followed instead of the 1980 version because it provides more appropriate guidance for testing the HEPA filters. Per ANSI N510-1989 paragraph 8.2.2, the upstream HEPA filter bank air distribution test is adequate to verify uniform air distribution for the downstream adsorber if the banks have relatively uniform geometry and cross sectional area, and are not interrupted by turns or bends. These filter units meet this design criteria. Since ANSI N510-1989 is the latest version for testing of nuclear air treatment systems, the intent of these air distribution tests will be met, thereby maintaining the design basis function of the filter units

The procedure has also been revised to allow taking DOP samples from the exterior of the carbon filter unit during the air aerosol mixing uniformity test. New DOP testing ports were added upstream of the HEPA filter units by a minor modification. Since the modification was designed per ANSI N510-1980, the air aerosol mixing uniformity test will follow these guidelines.

Overall, the air distribution and air aerosol mixing uniformity tests as well as the HEPA and carbon bypass leakage tests will assure that the filter units are operable, per the requirements of NRC Regulatory Guide 1.52 as well as Technical Specification 3.9.3 and 5.5.11. The test acceptance criteria for the air distribution and air aerosol mixing uniformity tests meet or exceed the requirements in UFSAR Table 12-28.

The option to install multiple sets of spargers for DOP and R-11 injection to Containment Purge Filter Unit-2A was also added. Multiple spargers may be needed to ensure air aerosol is uniformly distributed through the ducting prior to entry into the filter unit. The procedure was also revised to perform an air distribution test for the Containment Purge Filter Unit-2B HEPA filter bank. The air distribution test is being performed on Containment Purge Filter Unit-2B to determine if air flow through the filter units is balanced.

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, as well as deletion of U-tube manometers, inclined manometers, and psychrometers as test equipment. Air Data Multimeters are now used to perform these testing functions. Procedure PT/2/A/4450/001D was also administratively revised to the ensure that the Incore Instrument Filter Unit 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 Engineering . Additional equipment added to the procedure includes a new DOP generator and test equipment necessary for the air distribution and air aerosol mixing uniformity tests. All testing equipment including the new DOP generator meets the requirements of ANSI N510-1980 and ANSI N510-1989.

The Containment Purge Ventilation System exhaust will only be temporarily configured to support testing. The procedure has 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: Innage testing of the Containment Purge Ventilation System filter units 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 will be opened to maintain Containment Isolation Valves in the closed position. Control Room selector switches and push buttons for Containment Isolation Valves will be maintained in the "Blocked Closed" position. Control power will be removed from Motor Control Center 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 are not exceeded. The filtered exhaust backdraft damper 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 FHA 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 and will not adversely affect the Containment Purge Ventilation System as described in the UFSAR. After testing 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.

Changes have been made in the testing procedure to allow testing guidelines of ANSI N510-1989 to be used in place of ANSI N510-1980. The updated revision of ANSI N510 provides improved guidelines for air distribution tests. UFSAR Table 12-28 will be revised to reflect use of ANSI N510-1989 in place of ANSI N510-1980. Innage testing of filters will continue to 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.

A 10CFR 50.59 evaluation concluded that this activity can occur without prior NRC approval. No Technical Specification changes are required. UFSAR Table 12-28 will be revised.

55 **Type:** Procedure **Unit:** 1

Title: Temporary Test Procedure TT/0/A/9300/030A

Description: Temporary Test Procedure TT/0/A/9300/030A will be completed for Control Room Area Chiller Train A cooling water pressure switch 0YCPS6080. During the test, instrument tubing to the differential pressure switch will be opened. This will allow air to enter the pressure switch. During testing, instrumentation will be utilized to assist engineering in evaluating potential changes to the pressure switch setpoint. The opposite train must be fully operable and in service during the testing. Control wiring leads from the chiller will be opened and connected to circuitry that will measure and record pressure response on a local test computer. The chilled water pump will be operated as often as necessary to provide flow through the chiller evaporator. Introduction of air into the system should create high initial pressure transients at the differential pressure switch. These pressure transients will be monitored with operation of the chilled water pump.

Evaluation: The design basis of the Control Room Area Ventilation/Chilled Water System is to ensure that the control room will remain habitable for Operations personnel during and following all credible conditions and to ensure that the ambient temperature does not exceed the allowable temperature for the continuous duty rating of the equipment and instrumentation cooled by the system.

These functions are accomplished by pressurizing the Control Room (greater than 0.125 inwc) with respect to all adjacent areas, filtering the outside air used for pressurization, and 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 90 degrees F.

The proposed testing will be completed only when the Train of Control Room Area Ventilation/Chilled Water System is out of service for maintenance. During the proposed testing, test instrumentation will be used to measure pressure transients. At the conclusion of the proposed testing, a separate performance test (PT/0/A/4450/008E) will be completed to ensure that the chiller is performing properly and meets its design basis.

There are no unreviewed safety questions associated with this procedure. This activity will have no effect on any accidents analyzed in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required.

101 Type: UFSAR Change **Unit:** 0

Title: Change to Selected Licensee Commitments 16.7-10 "Radiation Monitoring for Plant Operations" and 16.11-7 "Radioactive Gaseous Effluent Monitoring Instrumentation"

Description: Selected Licensee Commitments 16.7-10 "Radiation Monitoring for Plant Operations" and 16.11-7 "Radioactive Gaseous Effluent Monitoring Instrumentation" are being revised. The revision concerns Radiation Monitor EMF39 (Containment Atmosphere High Gaseous Radioactivity - Low Range) to allow for gaseous radioactive releases to continue via the Containment Purge Ventilation System in the event that Radiation Monitor EMF39 becomes inoperable with the affected unit outside of Modes 1, 2, 3, and 4; provided that Radiation Monitor 1EMF36 (Unit Vent High Gaseous Radioactivity - Low Range) is operable and in service.

Selected Licensee Commitment 16.7-10, Table 16.7-10A, Action C currently states:

"With less than the minimum channels operable requirement, operation may continue provided the Containment Purge and Exhaust Valves are maintained closed."

Action C is being changed to read as follows:

"With less than the minimum channels operable requirement, operation may continue with EMF39 inoperable for up to twelve (12) hours with the affected unit outside of Modes 1, 2, 3, and 4, provided the following conditions are satisfied:
(a) EMF36 for the affected unit is operable and in service, and
(b) the Reactor Coolant System for the affected unit has been vented and,
(c) if the reactor vessel head is not in place (bolts not required) one of the following two conditions must be met:
(c1) all irradiated fuel assemblies have been removed from the containment structure, or
(c2a) the lifting of heavy loads over the reactor vessel have been suspended, and
(c2b) movement of irradiated fuel assemblies within the containment has been suspended
Otherwise maintain the Containment Purge and Exhaust Valves closed."

Selected Licensee Commitment 16.11-7, Table 16.11-5 "Radioactive Gaseous Effluent Monitoring Instrumentation", Action F currently states:

"With the number of channels operable less than required by the minimum channels operable requirement, immediately suspend purging of radioactive effluents via this pathway."

Action F is being changed to read as follows:

"With the number of channels operable less than required by the minimum channels operable requirement, operation may continue with EMF39 inoperable for up to twelve (12) hours with the affected unit outside of Modes 1, 2, 3, and 4, provided the following conditions are satisfied:
(a) EMF36 for the affected unit is operable and in service, and

(b) the Reactor Coolant System for the affected unit has been vented and,
(c) if the reactor vessel head is not in place (bolts not required) one of the following two conditions must be met:
(c1) all irradiated fuel assemblies have been removed from the containment structure, or
(c2a) the lifting of heavy loads over the reactor vessel has been suspended, and
(c2b) movement of irradiated fuel assemblies within the containment has been suspended

Otherwise, immediately suspend purging of radioactive effluents via this pathway.

To avoid confusion and reduce the potential for human error, limitation (b) above has been included in both Action C and Action F. Technically, Action C does not require limitation b.

Evaluation: It was concluded that there is no increase in the probability of accidents previously evaluated or unevaluated in the SAR. This change will not affect the ability of systems, structures or components to perform their function. The consequences of an accident previously evaluated in the SAR have also not been changed from the perspective that the analyses performed in the preparation of the SAR did not credit isolation of containment following a dropped fuel accident, thereby all activity released into the containment was assumed to eventually reach the environment.

It is recognized that there is a possibility of radioactive material being released to the environment as a result of this activity. The activity immediately released to the environment (during the period of EMF39 inoperability) is estimated to be a maximum of 680 curies of noble gases (iodines and particulates are assumed to be filtered via the Containment Purge Ventilation System Filter Train(s) in service). The activity released during a Fuel Handling Accident is estimated to be 8.34 E+05 Curies. The radiological whole body dose for the Fuel Handling accident to members of the public was calculated to be 0.59 Rem and 0.019 Rem at the exclusion area boundary and low population zone, respectively (UFSAR Table 15-47). The 680 Curies of activity potentially released during EMF39 inoperability is such a small fraction (0.081%), it is estimated that the dose to the public would be proportionally smaller (0.48 MR and 0.02 MR, at the Exclusion Area and Low Population Zone, respectively). This increase in exposure is within the guidance defining a minimal increase in risk.

Radiation Monitors are not accident initiators, therefore revising the Selected Licensee Commitments affecting this radiation monitor does not result in more than a minimal increase in the frequency of an accident previously evaluated in the UFSAR.

The revision to the Selected Licensee Commitments concerning Radiation Monitor EMF39 (Containment Atmosphere High Gaseous Radioactivity - Low Range) addresses under what conditions the Radiation Monitor can be inoperable, and still allow the release of effluents from the containment structure. This radiation monitor is not nuclear safety-related. The UFSAR takes no credit for this monitor being able to perform its function during and/or following a design basis accident. Therefore, this activity does not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR.

However, the proposed activity would allow releases of potentially radioactive gases from the containment structure without being quantified and controlled by Radiation Monitor EMF39. There are several criteria that must be met in order to enable this proposed action. First, is the requirement that Radiation Monitor EMF36 be operable and in service for the affected unit. Radiation Monitor EMF36 does not perform an automatic isolation of the Containment Purge Ventilation System pathway when it senses high radiation. In this respect it is not an equivalent monitor. However, Radiation Monitor EMF36 will alarm should significant increases in radioactivity occur (Fuel Handling Accident) alerting operators of a significant change in radiological effluents being released to the environment. Releases from containment with both Radiation Monitor EMF36 and Radiation Monitor EMF39 inoperable constitute unmonitored releases and should be prevented.

Second, is the requirement that the Reactor Coolant System (RCS) has been vented. This requirement ensures that the RCS contains a minimum concentration of noble gases. Further the amount of noble gas activity released due to a RCS leak inside containment will be conservatively bounded by the assumptions made in calculation CNC-1227.00-00-0060, "OAC 'Delta Count Rate' Setpoint Calculation for Radiation Monitor EMF-39 and Radiation Monitor EMF-38 to Comply with Reactor Coolant Leakage Detection Requirements." Even though this calculation applies to Modes 1 through 4, the conclusions made on noble gas concentration are conservative when applied to plant conditions outside Mode 1 through Mode 4.

Third, is the requirement that the irradiated fuel present is protected from damage. This ensures that the probability of a Fuel Handling Accident is minimized.

This evaluation includes a spreadsheet used to estimate the build up of noble gasses resulting from an unidentified one gallon-per-minute leak inside the containment structure. The inputs to this spreadsheet are based upon calculation CNC-1227.00-0060 (OAC 'Delta Count Rate' Setpoint Calculation for EMF-39 and EMF-38 to Comply with Reactor Coolant Leakage Detection Requirements, revision 2), UFSAR Chapter 9.4.5 (Containment Purge Ventilation System), UFSAR Table 11.2 (Design Basis Reactor Coolant Radioactivity Concentrations) UFSAR Chapter 15.7 (Radioactive Release From a Subsystem or Component), UFSAR Table 15-45 (Activities in Highest Inventory Discharged Assembly for Postulated Fuel Handling Accidents), and UFSAR Table 15-47 (Parameters for Postulated Fuel Handling Accident Inside Containment).

Although calculation CNC-1227.00-0060 is based on the unit being at power, its use in this application establishes a conservative boundary in the assumption of the quantities of material released when the unit is outside of Modes 1, 2, 3, or 4. Since Radiation Monitor EMF39 is a noble gas monitor, the portion of the calculation affecting particulate and iodine concentrations were omitted for this evaluation.

Calculation CNC-1227.00-0060 determined that 9.8E+05 microCuries of Xe-133 equivalent activity is introduced to the lower containment atmosphere every minute. Being a gas, it is assumed that this activity instantly disperses throughout the lower containment volume. During Modes 1, 2, 3, and 4 this activity would accumulate; but since the proposed activity is exclusive of these modes, it is assumed that the

Containment Purge Ventilation System is in service and is sweeping out a portion of this activity from the containment structure and releasing it to the environment. After twelve (12) hours, the estimate of activity released to the environment is 680 Curies of noble gases. The period of twelve (12) hours was selected by Radiation Protection Management as the length of a typical shift. It may be possible to operate for a longer period of time without Radiation Monitor EMF39 inservice, however this was the administrative restriction placed on this proposed activity.

UFSAR Table 15-45 provides a listing of the noble gas activity released during a Fuel Handling Accident. The noble gas portion of the gap activity is 8.34E+05 Curies. The estimation of 680 Curies of noble gases released during the 12-hour period of Radiation Monitor EMF39 inoperability represents 0.081% of the Fuel Handling Accident consequences. Applying this percentage to the whole body dose estimates provided in UFSAR Table 15-47 results in the dose to the public being proportionately smaller (e.g. 0.48 millirem and 0.02 millirem, at the Exclusion Area Boundary and Low Population Zone, respectively). This increase in exposure is within the guidance defining a minimal increase in risk.

A 10CFR50.59 evaluation concluded that this change could be made without prior NRC approval. No Technical Specification changes are required. UFSAR changes are required for the Selected Licensee Commitments 16.7-10 "Radiation Monitoring for Plant Operations" and 16.11-7 "Radioactive Gaseous Effluent Monitoring Instrumentation.".

102 **Type:** UFSAR Change **Unit:** 0

Title: Change to Selected Licensee Commitments 16.7-11 "Position Indication System-Shutdown"

Description: Selected Licensee Commitment 16.7-11 (Position Indication System- Shutdown) is being revised to change the testing requirements. The existing testing requirement states "Testing requirements are specified in Technical Specification Surveillance Requirement 3.1.7.1. Technical Specification 3.1.7 is only applicable in Modes 1 and 2. Selected Licensee Commitment 16.7-11 is only applicable in Modes 3, 4, and 5. This leads to confusion about the actual testing requirements for Modes 3, 4 and 5. The correct testing requirements have been previously determined by Engineering and Regulatory Compliance.

Engineering documented all of the testing that had been completed on DRPI. This was reviewed by Regulatory Compliance, and it was agreed that the testing was sufficient for Modes 3, 4, and 5. Regulatory Compliance determined that Surveillance Requirement 3.1.7.1 is not applicable in the modes of applicability of this Selected Licensee Commitment. The basis for that determination is summarized below:

Selected Licensee Commitment 16.7-11 was formerly Technical Specification 3.1.3.3. When Technical Specification 3.1.3.3 was converted to Selected Licensee Commitment 16.7-11 during the Improved Technical Specification conversion process, the justification was that position indication is not required per the Technical Specification in lower modes because there is no potential for a power distribution anomaly. This is documented in the Improved Technical Specification submittal. The +/- 12 step alignment is not assumed when in Modes 3, 4 and 5, since no reactor power is being generated and the reactor is subcritical. The rod alignment and position indication is only necessary when the reactor is critical. Therefore, it follows that since Digital Rod Position Indication is not required for the purpose of detecting a power distribution anomaly in lower modes that Surveillance Requirement 3.1.7.1 is not applicable in the lower modes.

Technical Specification Surveillance Requirement 3.1.7.1 is applicable in Modes 1 and 2, and is performed during rod withdrawal as part of the rod drop timing procedure. This satisfies the frequency of "once prior to criticality after each removal of the reactor head."

The testing requirements for Selected Licensee Commitment 16.7-11 are being revised to state:

The position indication system shall be demonstrated operable after each removal of the reactor head by

1. Performing a power supply calibration of the digital rod position indicators and
2. Performing a functional test of the digital rod position indicators

Performing these tests will verify that DRPI is working correctly. The functional test, using a coil stack simulator, verifies that the DRPI indications are correct for each rod from 0-228 steps. Also, the functional test verifies that correct/valid position data is received for each control rod when its respective head cable is connected. This testing verifies that DRPI is working properly.

Evaluation: This activity does not change any plant systems, structures, or components. The DRPI System is not an accident initiator. This change has no effect on any accident evaluated in the SAR. No Technical Specification changes are required. A UFSAR change will be made to Selected Licensee Commitment 16.7-11.

111 Type: UFSAR Change **Unit:** 0

Title: Conversion of the Selected Licensee Commitments (SLC) Manual to Improved Technical Specification (ITS) Format

Description: The purpose of this activity is to convert the Selected Licensee Commitments Manual to the same format used by the Catawba Improved Technical Specifications. It has been determined that having both the Selected Licensee Commitments Manual and the Technical Specifications in the same format will provide benefit to the plant operating staff. Throughout the conversion process, technical changes to the Selected Licensee Commitments Manual were kept to a minimum. Where existing Selected Licensee Commitments requirements could obviously be improved upon from a technical perspective, then changes were made as necessary. All of the changes to the Selected Licensee Commitments Manual are justified below.

Changes in the Selected Licensee Commitments (SLC) Manual associated with conversion of the Manual to Improved Technical Specification (ITS) format.

Administrative changes:

A1. The SLC Manual was reformatted to present the information in a style and nomenclature that is consistent with that used in the ITS. Minor changes were made to the SLCs to incorporate ITS format, terminology, references, numbering or editorial preference necessary to convert the SLC Manual to the ITS format and convention. These changes are considered administrative and do not affect any technical requirements specified in any SLC.

A2. For SLC 16.6-1 (Engineered Safety Features-Containment Sump), information presently contained in the testing requirements section that is of a descriptive nature is being relocated to the bases section. This information consists of examples of debris types (rags, trash, clothing, etc.), as well as the consequences of the debris being transported into the containment sump. Relocation of this descriptive information from one SLC section to another does not change the requirements of the SLC, therefore the change is considered administrative.

A3. For SLC 16.6-4 (Engineered Safety Features-Chlorine Detectors and Associated Circuitry), there is a temporary footnote pertaining to implementing modification work associated with deleting the high chlorine automatic control room intake isolation function and reclassifying the chlorine detectors and/or associated circuitry to a non nuclear safety related status. The note was deleted because it was no longer required since the modification work has been completed.

A4. For SLC 16.8-1 (Electrical Power Systems-Containment Penetration Conductor Overcurrent Protective Devices), the remedial action option of restoring the inoperable equipment to operable status was deleted. Restoring inoperable equipment to operable is always an option for compliance with SLC requirements and does not need to be stated. This change is considered administrative.

A5. The applicability of SLC 16.9-6 (Auxiliary Systems-Fire Protection Systems-Fire

Detection Instrumentation) is expanded to include the phrase "or is in service". Certain equipment in SLC Table 16.9-3 (Fire Detection Instruments) (e.g. the Containment Auxiliary Carbon Filter Units in fire zones 177, 178, 179, and 180) are not nuclear safety related and do not have any operability requirements according to the Technical Specifications or the Selected Licensee Commitments. Hence, the existing applicability has no meaning for these components. This change does not impose or remove any technical requirements for any equipment and is therefore considered administrative.

A6. For SLCs 16.9-8 (Auxiliary Systems-Boration Systems Flowpaths-Operating) and 16.9-10 (Auxiliary Systems-Boration Systems Charging Pumps-Operating), an obsolete footnote was deleted. The activities addressed by the footnote (Nuclear Service Water System pipe cleaning) have already been completed. The change does not cause any technical modification to the SLC and is therefore considered administrative.

A7. Several administrative changes were made to SLC 16.11-15 (Radiological Effluent Controls Interlaboratory Comparison Program) regarding descriptive material associated with the program. These changes do not alter any technical requirements of the program and have been justified in a 10CFR50.59 screening prepared by Radiation Protection.

A8. In SLC 16.11-20 (Radological Effluents Controls-Explosive Gas Monitoring Instrumentation), a 30 day time period was specified for the preparation and submittal of a Special Report when the explosive Gas Monitoring Instrumentation channel(s) are not restored to operable status within the required completion time. No time period is currently specified for preparation and submittal of this report. Thirty days is consistent with the time frame for submitting other types of Special Reports. This change is considered administrative.

Less Restrictive Changes:

L1 For SLC 16.2 (Applicability), this less restrictive change provides for a justified deviation from the requirements of a commitment under extenuating circumstances, as allowed by the Station Manager or his designee.

L2. For SLC 16.8-1 (Electrical Power Systems-Containment Penetration Conductor Overcurrent Protective Devices), this less restrictive change deletes the Containment Penetration Containment Overcurrent Protection Devices for the Incore Instrument Tunnel Booster Fan Motor 1A and 1B. The justification for this deletion is provided in a 10CFR50.59 evaluation prepared by Engineering for Modification CE-60316 (a summary of this evaluation was included in the 1998 10CFR50.59 Annual Summary Report).

L3. Selected Licensee Commitment 16.7-10 (Instrumentation Radiation Monitoring for Plant Operations) Remedial Action D is applicable to the case where the number of operable channels is one less than the minimum number of channels operable requirement. Table 16.7-10A specifies the minimum channels operable as 2 (1 per intake). Under the existing SLC, there is no condition applicable when both radiation monitor EMF-43A and EMF-43B are inoperable simultaneously. A new Condition D was added to state, "One Control Room Air Intake-Radiation Level-High Gaseous Radioactivity channel inoperable in one or both Control Room Intakes". Since the new condition is applicable to one or both radiation monitors inoperable, it provides a condition that does not exist under the current SLC, and is therefore considered less restrictive by providing this additional flexibility (refer to the discussion of change M2).

L4. For SLC 16.9-8 (Auxiliary Systems-Boration Systems Flowpaths-Operating), SLC 16.9-10 (Auxiliary Systems-Boration Systems Charging Pumps-Operating), and SLC 16.9-12 (Auxiliary Systems-Boration Systems Borated Water Sources-Operating), the existing requirement of placing the unit in Mode 5 within thirty hours is being modified to only require placing the unit in Mode 4 with any Reactor Coolant System cold leg temperature less than or equal to 285 degrees F., which will place the unit outside of the modes of applicability of the SLC. The applicability of the subject SLC is Modes 1, 2, and 3 and Mode 4 with Reactor Coolant System cold leg temperatures greater than 285 degrees F. Therefore, it is not necessary to require the unit to be placed in Mode 5 in order to exit the modes of applicability of the SLC. This change is considered less restrictive.

L5. For SLC 16.9-22 (Auxiliary Systems Control Room Area Ventilation Systems-Intake Alarms), the existing remedial actions are replaced with new actions that more closely correspond with the remedial actions associated with inoperable control room high radiation monitors and smoke detectors in SLC 16.7-10 (Instrumentation Radiation Monitoring for Plant Operations) and SLC 16.9-6 (Auxiliary Systems-Fire Protection Systems-Fire Detection Instrumentation) respectively. Having an inoperable Control Room Area Ventilation System Intake Alarm for radiation or smoke monitoring is no worse than having either the radiation monitor or smoke detector itself inoperable. The remedial actions associated with inoperable control room high radiation monitors or smoke detectors do not require shutting down the affected unit(s) or the suspension of core alterations or positive reactivity changes. Hence, there is no basis for requiring these actions when an intake alarm becomes inoperable. The new remedial action for inoperable radiation intake alarms is consistent with that for an inoperable Control Room Air Intake-Radiation Level-High Gaseous Radioactivity channel. The new remedial action for inoperable smoke intake alarms is consistent with that for inoperable fire detection instrumentation. This change is considered less restrictive.

L6. For SLC 16.7-10 (Instrumentation Radiation Monitoring for Plant Operations), the five gallon per day setpoint for the Nitrogen-16 monitors (Radiation Monitors EMF-71, EMF-72, EMF-73 and EMF-74) is being changed to specify that the setpoint will be as required by the primary to secondary leak rate monitoring program. The actual setpoint may be varied depending upon plant conditions in accordance with Duke Nuclear System Directive 513, which governs the monitoring program. The justification for this change was provided in a separate 10CFR50.59 evaluation developed by Engineering. This change provides additional plant flexibility and is therefore considered less restrictive.

L7. For SLCs 16.13-2 (Conduct of Operations-Technical Review and Control) and 16.13-3 (Conduct of Operations-Operating Plant Operations Review Committee), changes are being proposed related to qualifications for personnel performing procedure reviews and also allowing proposed Technical Specification changes that are administrative to be exempted from full Plant Operations Review Committee (PORC) and Nuclear Safety Review Board (NSRB) review. Also, a number of other editorial changes to these SLCs are proposed. A separate justification per the 10CFR59.59 process was developed by the Duke General Office Regulatory and Industry Affairs associated with these changes.

L8. For SLC 16.7-6 (Instrumentation-Nuclear Service Water System Discharge Instrumentation) the remedial actions were rewritten to be more appropriate for the nature and function of the Nuclear Service Water discharge instrumentation. The revised remedial actions state that with one or more instrument loop(s) and or annunciators inoperable, either the operator can immediately ensure that the associated Nuclear Service Water trains(s) are not in service or the affected Nuclear Service Water loop(s) discharge(s) are aligned to the Standby Nuclear Service Water Pond. The proposed remedial actions will ensure that there will not be a scenario where operable Nuclear Service Water discharge instrumentation will not be available to alert operators to a loss of the non nuclear safety related Lake Wylie Discharge. The revised remedial actions also allow 72 hours to restore the inoperable instrumentation to operable status. Seventy two hours is considered to be a reasonable amount of time to allow for inoperability of the affected instrumentation, since Technical Specification 3.7.8 for the Nuclear Service Water System currently allows 72 hours for an inoperable Nuclear Service Water train. It is not necessary to specify a direct entry into Technical Specification 3.7.8 or 3.0.3 as is currently specified in the SLC, because Technical Specification 3.7.8 already specifies the appropriate Required Actions for inoperable Nuclear Service Water System trains. This change is considered less restrictive.

More restrictive changes:

M1. This change adds a 72 hour time limit for performing an engineering evaluation to determine the effects of the out-of limit condition on the structural integrity of the pressurizer and for determining that the pressurizer remains acceptable for continued operation. No time limit is presently given in SLC 16.5-4 (Reactor Coolant System-Pressurizer) in conjunction with these activities; therefore, the change is considered more restrictive. Seventy two hours is an appropriate time limitation for these activities since it is consistent with the general time frame for performing operability evaluations and is also consistent with Technical Specification 3.4.3 (Reactor Coolant System Pressure/Temperature Limits). Technical Specification 3.4.3, Required Action A.2 specifies a 72 hour completion time for determining that the RCS is acceptable for continued operation when P/T limits are exceeded.

M2. Current SLC Remedial Action 16.7-10 (Instrumentation-Radiation Monitoring for Plant Operations), Remedial Action D requires initiating and maintaining, within one hour, operation of one train of the Control Room Area Ventilation System with flow through the HEPA filters and activated carbon adsorbers. At Catawba, the Control Room Area Ventilation System always operates in the filtered mode; therefore, the discussion pertaining to flow through the HEPA filters and activated carbon adsorbers is being deleted. New Required Action D.1. is written to require immediately initiating action to restore the inoperable channels to operable status. This action is acceptable, since the filtered operation of the Control Room Area Ventilation System provides the required operator protection during an accident in the event that one or both radiation monitors are inoperable. Since the present SLC requirement does not mandate restoration of an inoperable radiation monitor, this change is considered more restrictive. The requirement to initiate and maintain the Control Room Area Ventilation System operation within one hour is retained as required action D.2. (refer to discussion of change L3).

Evaluation: While these Selected Licensee Commitment changes involve Selected Licensee Commitments pertaining to a number of plant structures, systems, and components (SSCs), in all cases the changes are allowed to be made under the provisions of 10 CFR 50.59. No SSC is modified or caused to operate in a different manner from current operation. No seismic, environmental, materials, or reactivity effects are created on these SSCs as a result of the proposed Selected Licensee Commitment changes. No new failure modes or effects or new types of system/component interactions will be introduced upon any SSC as a result of these changes. SSC behavior during steady state and transient conditions will be unaffected by any of these Selected Licensee Commitment changes. Plant response to any external phenomena, natural or man-made, likewise will not be affected. No plant transient or accident analyses will require revision as a result of these changes. No actual testing requirements are being modified in a significant manner by these proposed changes. No actual regulatory commitments are being eliminated or reduced by these changes.

91 Type: UFSAR Change **Unit:** 0

Title: Selected Licensee Commitment Changes to SLC Table 16.7-10A, Table 16.7-10B, SLC 16.11-2, SLC 16.11-3, SLC 16.11-5, SLC 16.11-6, TS Bases B3.4.15 and TS SR 3.4.15.2

Description: Selected Licensee Commitments (SLCs) 16.7 -10 (Tables 16.7-10A and 16.7-10B) which addresses Radiation Monitoring for Plant Operations, SLC 16.11-2 which addresses Radioactive Liquid Effluent Monitoring Instrumentation, SLC 16.11-3 which addresses Radiological Dose, SLC 16.11-5 which addresses Chemical Treatment Ponds, SLC 16.11-6 which addresses Radiological Gaseous Effluents, and Technical Specification Bases B3.4.15 and SR 3.4.15.2 which address RCS Leakage Detection Instrumentation are being revised. Most of these revisions are clarifications of an editorial nature.

The change that is not editorial in nature is the change to the requirements for when Radiation Monitor EMF39 (Containment Atmosphere High Gaseous Radioactivity (Low Range)) is required. In SLC 16.11-7 (SLC Table 16.11-6, Item 4) "Modes for which Surveillance is Required" presently states "at all times" and Item 5 presently states "at all times except when the isolation valve is closed and locked." This is illogical since Item 4 relates to the Containment Purge System which is locked closed during Modes 1 through 4 and Item 5 relates to the Containment Air Release and Addition System whose valves are routinely cycled open and closed in order to relieve containment pressure during Modes 1 through 4. Further, the existing restriction (Item 4 is presently "required at all times" and Item 5 is presently "required at all times except when the isolation valve is closed and locked") is contradictory to the UFSAR and Technical Specifications related to the operation of the Containment Purge System and Containment Air Release and Addition System. This will be fixed by changing the "Channel Operational Test" Column entry for the Containment Purge System from Q (Quarterly) to "R" (Refueling Outage) frequency and under the "Modes for which Surveillance is Required" column, deleting the phrase "Modes 5 and 6".

Evaluation: Since EMF39 is not an accident initiator, these changes have no effect on the frequency of occurrence of any of the accidents previously evaluated in the UFSAR. A 10CFR50.59 evaluation determined that these changes could be made without prior NRC approval. No Technical Specification changes are required. No UFSAR changes are required.

33 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Figure 7-14

Description: This UFSAR change will replace the functional logic diagram Figure 7-14. The current diagram, while technically accurate, does not serve to clarify how the circuits function. The replacement diagram has been arranged to clarify various text section discussions of the control circuits by adding several component symbols, revising several symbols and altering the way the logic is presented so that it more closely follows the descriptive material.

This change will revise the logic diagram for the circuits which are controlled by the Containment Pressure Control System (CPCS). This diagram is used in conjunction with various UFSAR textual sections, and other design documents to provide an understanding of the containment pressure and temperature control mechanisms. This diagram does not represent either a connection drawing or an electrical elementary, rather, it demonstrates the relationship of components to one another. Logic diagrams are used in several UFSAR locations for the same purpose, and the changes to this diagram are to show the relationships in the same manner as the other diagrams do.

The CPCS has a set of components called Alarm Modules, which invert one of the functions of the pressure transmitters. This function is to block actuations when the pressure drops below the actuating set point. The modules are not shown on the diagram, but their logic function is. This change makes use of AND gates to show how this function relates to the operation or shut-down of components so the reader does not need to mentally invert the signals as must be done to understand the current diagram.

Four symbols are being added, two timers, a differential pressure transmitter, and a manual control symbol. These functions are not currently shown in the diagram, and their inclusion serves to increase the clarity of the overall function. An individual studying these systems would become aware of the existence of these components and would find the diagram confusing if the function is not shown.

The symbol for the fan has been altered to show that the control mechanism is completely different from the controls for the spray pump, which is not apparent from the current diagram. When comparing the diagram to electrical elementaries, it becomes apparent that the fan motors do not have a separate stop circuit as the pumps do. This change clarifies the start/stop function for the two different cases.

The current drawing does not provide any indication for how the dampers are closed, and this change provides a manual action symbol to show this function.

Evaluation: This diagram change does not alter any plant components, nor does it alter any plant circuitry. There are no errors in the plant drawings. This change merely alters the way several circuits are represented on a functional diagram used to help readers understand the relationship of the several control circuits. There are no unreviewed safety questions associated with this change. No Technical Specification changes are required. UFSAR Figure 7-14 will be revised.

72 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 11.5.1.2

Description: It has been identified that the design basis specification for the Radiation Monitoring System, states that EMF43A and EMF43B (Control Room Air Intake Monitors) are process monitors that sample the Control Room Area Ventilation System. These monitors provide no control functions, serving only to notify Control Room personnel of increasing radiation levels in the affected air intake."

Contrary to this statement, UFSAR Section 11.5.1.2, System Description states in part, ".. some of the monitors (e.g., Control Room Air Intake) perform control functions during postulated accident conditions."

The statement that the Control Room Air Intake monitors perform control functions is incorrect. Therefore the UFSAR is being revised to delete that statement.

To further support this change, the UFSAR later states, in Section 11.5.1.2.2.5, "Control Room Air Intake Monitors" that "..gaseous activity in a control room air intake which exceeds a preset limit actuates an alarm in the control room. Operations can then isolate the affected intake if desired."

Evaluation: Removal of this erroneous information (that the Control Room Air Intake monitors perform a control function) does not increase the probability or consequences of accidents evaluated in the UFSAR. Similarly, removal of this erroneous information does not affect the equipment or any malfunction of the equipment. The removal of this erroneous information does not create the potential for accidents or malfunctions not analyzed in the UFSAR. The removal of this erroneous information does not affect exposures as outlined in 10CFR100. Therefore this UFSAR change can be made without prior NRC Approval. No Technical Specification changes are required. UFSAR Section 11.5.1.2 will be revised.

16 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 12.3.2.1

Description: UFSAR Section 12.3.2.1 is being revised to indictae that the computer codes listed (ANISN, SABINE, KAP IV, SHIELD, DOT, and N23BURP) were codes that were used for initial analysis of the Primary Shield and Auxiliary and Reactor Building Shields. The change also adds information about the presently used shields, (MCNP, SCALE, and QAD-CGGP-A). The revision of the status of the initial licensing codes is an editorial change. MCPN, SCALE, and QAD-CGGP-A are widely used codes in the nuclear industry. These sheilding codes are recognized by the NRC as the appropriate codes to be used for radiation shielding analysis.

Evaluation: There are no unreviewed safety questions associated with this UFSAR change. This change does not affect plant activities. Computer codes are not initiators of any accidents analyzed in the UFSAR. The change will have no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. UFSAR Section 12.3.2.1 will be revised.

77 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 3.11, "Environmental Design of Mechanical and Electrical Equipment"

Description: UFSAR Section 3.11 is being revised to remove outdated information and capture updated licensing basis information related to environmental qualification requirements for operation of accident mitigation equipment in normal and post-accident environments. Duke Power Company's response to NUREG 0588 has been superseded by Nuclear System Directive-303, which is the governing environmental qualification program document for the company. Other subtier documents, which define the implementation of the program, have also been added to the UFSAR.

Evaluation: This activity has no safety significance. UFSAR Section 3.11 contains outdated information related to environmental qualification of equipment. This section references Duke Power Company's response to NUREG 0588 as the Company's program for Environmental Qualification of equipment. This response has been revised over the years and the Environmental Qualification Program is now governed by NSD-303. The major subtier guidance documents which control the implementation of the Environmental Qualification Program are: 1) the Environmental Qualification Criteria Manual, which defines the environmental conditions both inside and outside of the reactor building following a postulated design basis event, 2) the Equipment Database, which identifies the equipment requiring environmental qualification, and 3) the Environmental Qualification Maintenance Manual (EQMM-1393.01), which defines the requirements for maintaining the environmental qualification of the accident mitigation equipment for the life of the plant.

A 10CFR50.59 evaluation determined that this UFSAR change can be made without prior NRC approval. No Technical Specification changes are required. UFSAR Section 3.11 will be revised.

76 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Section 9.3.2.2.1

Description: UFSAR Section 9.3.2.2.1 which addresses the Nuclear Sampling System is being revised. The revision will change the wording of the section to allow both the upper shell and blowdown sample lines of any steam generator to be open simultaneously. Currently, the UFSAR states: "Containment isolation valves are provided in each of the steam generator sample lines and are arranged so that the operator may select sample flow from either the upper shell or the blowdown line of each steam generator." This revision will reword this portion of Section 9.3.2.2.1 to state: "Containment isolation valves are provided in each of the steam generator sample lines and are arranged so that the operator may select sample flow from the upper shell and/or the blowdown line of each steam generator."

This change is required to provide adequate sample flow to the secondary and primary labs when any steam generator is not under pressure. During certain outage periods, steam generator samples are required to affirm the chemical composition of the contents of the steam generators. With only one sample valve open and no nitrogen overpressure on the generator, sample flow cannot be obtained with only one sample line open. With both lines open, upper shell and blowdown, the sample flow is sufficient to accommodate the sampling requirements.

Evaluation: The steam generator sample lines originating within containment are provided with remote, motor-operated containment isolation valves, both inside and outside containment, which are closed automatically by a containment isolation signal in the event of a LOCA. The isolation valves used for containment isolation of the process sampling lines are electric motor operated and fail "as is." These valves are used in groups for each penetration with the isolation valves inside containment supplied by one train of safety related power while the valve outside containment receives power from the other train of safety related power. Both interior and exterior valves receive appropriate automatic signals to close. Isolation of these lines is thus assured even with the assumption of a single failure.

The previous UFSAR statement only allowed one of the inside containment sample isolation valves to be open at a time. This revision will allow both of the inside containment valves to be opened. Should a containment isolation signal be generated in the event of a LOCA, either the safety related train valves inside or the single isolation outside of containment will secure this sample path. This alignment provides the same assurance of isolation as the existing revision in the event of a single failure.

There are no unreviewed safety questions associated with this UFSAR change. No Technical Specification changes are required. UFSAR Section 9.3.2.2.1 will be revised.

32 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Sections 6.7.1.3

Description: The UFSAR will be revised to correct information about the ice condenser floor drains. During design basis accidents, these drains permit melted ice and condensate to flow from the ice condenser to lower containment. During normal operation, the floor drain check valves are normally closed to minimize air leakage into or out of the ice condenser. The changes in wording add this function to the description of normal operation, changing the diameter of a small drain upstream of the floor drain check valve to agree with design documents, correcting the description of where the "transite" section is located in the drain line, and correcting the description of the use of sealant on the floor drain check valves to agree with procedures.

The twelve inch diameter ice condenser floor drains run from the floor of the ice condenser through the crane wall to lower containment. There are twenty ice condenser floor drains in each unit, each containing components with the same design, and having the same piping layout. Starting inside the ice condenser, each floor drain goes vertically downward through the floor, changes direction to horizontal using one or more 90 degree elbows, and extends through the Crane Wall into lower containment, where it ends with a floor drain check valve. The vertical section of the drain runs below the wear slab, and contains a transite section with low thermal conductivity. The bottom of the horizontal section that extends through the crane wall has a small diameter drain just upstream of the floor drain check valve. To minimize leakage, the interface where the valve seat meets the valve face is greased.

The design drawings that describe the SSCs are in agreement with the current UFSAR except for the following:

Section 6.7.1.3 of the UFSAR states that small drain line just upstream of each floor drain has a diameter of 1 1/2 inches. Design documents show the diameter of the small drain line to be 1/2 inch.

Section 6.7.1.3 of the UFSAR states that the "low conductivity (transite) section" is located vertically below the seal slab. This is the only instance where the words "seal slab" are used in connection with the ice condenser. Other text in the UFSAR (including text in Section 6.7.1.3), Figures in the UFSAR, and design drawings refer to the "wear slab". Figures in the UFSAR and design drawings show the ice condenser to be vertically below the wear slab.

Evaluation: There is no unreviewed safety question associated with this UFSAR change. These changes have no effect on the probability or consequences of accidents analyzed in the UFSAR. No Technical Specification changes are required. UFSAR Section 6.7.1.3 will be revised.

35 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment 16.11-7

Description: Selected Licensee Commitment 16.11-7 addresses Radioactive Gaseous Effluent Monitoring Instrumentation. Table 16.11-6 "Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements" lists the various affected instrumentation and has columns giving required frequencies for Channel Checks, source Checks, Channel Calibrations, and Channel Operational Tests. Also included in the table is a column which lists the operational modes for which the surveillance is required.

A problem was discovered with the information in this column for Table Row 4 "Containment Purge System" and Table Row 5 "Containment Air Release and Addition System."

Table Row 4 presently shows "**" under this column indicating "required at all times" and Table Row 5 is presently shows "*" under this column indicating "required at all times except when the isolation valve is closed and locked." This is incorrect since Item 4 relates to the Containment Purge System which is locked closed during Modes 1 through 4 and Item 5 relates to the Containment Air Release and Addition System whose valves are routinely cycled open and closed in order to relieve containment pressure.

The proposed change is to reverse these restrictions so that isolation of the Containment Purge System (which is locked closed during Modes 1 through 4) is "required at all times except when the isolation valve is closed and locked." Likewise, the isolation of the Containment Air Release and Addition System (whose valves are routinely cycled open and closed) is "required at all times."

Evaluation: This SLC revision will appropriately align the applicable modes for these surveillances to the actual operation of the systems and the associated containment ventilation isolation valves. Since the Technical Specification provided the guidance as to when the Containment Purge System containment isolation valves could be open (Modes 5, 6 and No Mode) and the changes are being made to the SLC to align it to the Technical Specification restrictions, there have been no changes to the accident initiators or behavior of safety systems or equipment vital to the operation of the plant.

There are no Unreviewed Safety Questions associated with this SLC change. No changes to the Technical Specifications are required. SLC 16.11-7 Table 16.11-6 will be revised.

28 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment 16.6-2 - Ice Bed Temperature Monitoring

Description: The operability requirement for the Ice Condenser Ice Bed Temperature Monitoring System is being deleted from the Catawba Nuclear Station Selected Licensee Commitment Manual.

Evaluation: It is not appropriate for operability of the Ice Bed Temperature Monitoring System to be a basis for establishing operability of the Ice Condenser System. The original Temperature Monitoring System operability requirement in the Westinghouse Standard Technical Specifications was not carried over into the Improved Technical Specification because it did not meet the criteria set forth in the Final Policy Statement. The former Technical Specification imposed a Limiting Condition of Operation (LCO), and a Surveillance Requirement that required twice daily Channel Checks, and Channel Functional Checks each refueling outage. The LCO predicated operability of the Temperature Monitoring System on 18 of its 48 sensors operating. The Action Statements ultimately required shutdown of the NSSS, because inoperability of the Ice Condenser System was inferred from the inoperability of the Temperature Monitoring System, if the Temperature Monitoring System could not be restored within the allowable outage time.

These operability and surveillance requirements were unduly conservative. Notwithstanding the fact that the Ice Condenser is a passive system, and that inoperability of the Ice Condenser System does not follow from inoperability of the Temperature Monitoring System, this Technical Specification imposed the same operability/surveillance requirements on the Ice Condenser System Temperature Monitoring System as is applicable to the Reactor Trip or Engineered Safeguards Features Actuation System Instrumentation. The ice bed itself meets Criterion 3 of the four criteria in 10CFR50.36 for a technical specification LCO, that were promulgated in an NRC Policy Statement. The Temperature Monitoring System, however, meets none of these criteria. Thus, when developing the Improved Technical Specifications, the industry determined that there was no safety significance involved in removing this Temperature Monitoring System operability requirement from the Technical Specification. For this requirement to reside in the Selected Licensee Commitments, with its Action Statement, that has been already determined unduly restrictive, and certainly more restrictive that the higher order Technical Specification is constraining, and is contrary to the intent of the NRC/Industry Technical Specification Improvement Program.

From the Ice Condenser System Technical Specification Bases, the two conditions that can degrade the ice bed are: the loss of ice by melting or sublimation:and the obstruction of flow passages through the ice bed due to buildup of frost or ice. The probability of occurrence of both of these degrading conditions is reduced by adherence to the provisions of the Improved Technical Specifications, which have been written in accordance with the intent of the Ice Condenser System design basis which is maintaining the ice bed at its design basis operating temperature, and minimizing air leakage into and out of the ice condenser. Verifying that the maximum temperature of the ice bed is <= 27 degrees F, per SR 3.6.12.1 (i.e., essentially within its design basis operating range of between 10 degrees F. and 29 degrees F.), ensures that the ice is kept below the melting point. Assurance of the existence of this operable condition in the ice bed is provided by means of a number of redundant and independent systems. The primary system is the Temperature Monitoring System. Secondary is the Temperature Monitoring System alarm system consisting of six temperature switches that independently monitor ice bed temperature, and alarm in the control room at a preset deviation from the limits of the ice bed temperature. This system is redundant to, and independent of the the Temperature Monitoring System and similarly serves to notify the control room of degradation of acceptable operating temperature conditions in the ice bed. To address temporary issues concerning the Operability of the Temperature Monitoring System, there is the very practicable alternative of lowering temperature sensors (e.g., RTDs) down from the Ice Condenser Intermediate Deck into the ice bed to provide temperature indication in the interim until the operability status of the Temperature Monitoring System has been restored.

There are other less direct means, inherent in the Ice Condenser System, to determine circumstances that would indicate, or potentially result in, degraded temperature conditions within the ice bed, e.g., alarm annunciation from the chiller units' glycol expansion tank level switches, the door position monitoring system, floor cooling system instrumentation, or backdraft damper switches. Rather than quantifying any temperature degradation within the ice bed, these tertiary systems can only provide evidence to the operator of conditions that could lead to an abnormal temperature within the ice bed. It is also unlikely that restoration of primary temperature indication would ever be delayed to the point where these systems would actually be relied upon to provide relative temperature status within the ice bed for any prolonged period. However, the existence of these secondary and tertiary means of indication serves to further substantiate the argument that declaring inoperability of the ice bed based solely upon the operability of the Temperature Monitoring System is inappropriate in that it potentially represents unnecessary risk to NSSS availability.

Thus, because the Selected Licensee Commitment LCO requirement is merely a constraint on the system that is one means used to demonstrate a state of compliance of a safety related SSC, and because Technical Specification compliance can be demonstrated by numerous means, the Selected Licensee Commitment requirement is unnecessarily conservative and redundant because the inoperability of the Temperature Monitoring System does not imply the inoperability of the ice bed. Therefore it is inappropriate for operability of the Temperature Monitoring System to be a basis for establishing the operability of the Ice Condenser System. Thus, in concordance with the objective of the industry Technical Specification Improvement Program and the resulting Improved Technical Specifications, the requirement of Section 16.2 of the Selected Licensee Commitment may be deleted, given this determination that there is no safety significance associated with doing so.

There is no unreviewed safety question associated with this revision to Selected Licensee Commitment 16.6-2. No Technical Specification changes are required. UFSAR Section 16.6-2 will be revised.

75 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment 16.6-4 and 16.9-22 to modify requirements for chlorine detectors

Description: Amendments 191/183 to the Catawba Operating Licenses were issued on June 28, 2001. These amendments delete Technical Specification requirements concerning chlorine detection/protection. Selected Licensee Commitments 16.6-4, Chlorine Detection Systems, and SLC 16.9-22, Control Room Area Ventilation System - Intake Alarms, are being revised commensurate with these license amendments. The Selected Licensee Commitment changes acomplish the following:

1. Editorially revise Selected Licensee Commitment 16.6-4 to change "chlorine detection systems" to "chlorine detectors and associated circuitry".2. Revise the Selected Licensee Commitment 16.6-4 Remedial Actions to allow a 30-day allowed outage time for one inoperable detector and/or circuitry in one or both intakes and to make the Remedial Actions consistent with the function of the detectors/circuitry. A statement was added that the provisions of Selected Licensee Commitment 16.2.3 are not applicable so that mode/condition changes are allowed with inoperable equipment. This is consistent with the identical Selected Licensee Commitment 16.2.3 exception contained in Selected Licensee Commitment 16.9-22.
3. Delete the Selected Licensee Commitment 16.6-4 12-hour channel check requirement from the Testing Requirements section. Also, delete the Testing Requirement requiring automatic isolation of the affected intake.
4. Add a reference to Selected Licensee Commitment 16.6-4 to the June 28, 2001 NRC letter issuing License Amendments 191/183.
5. Add a temporary footnote to Selected Licensee Commitment 16.6-4 to clarify that for the purposes of implementing modification work associated with deleting the high chlorine automatic control room intake isolation function and downgrading the chlorine detectors and/or associated circuitry to non-safety related status, entry into the Selected Licensee Commitment Remedial Actions is not required for pre-planned activities resulting in one inoperable detector and/or circuitry.
6. Modify the Bases for Selected Licensee Commitment 16.6-4 to expand the discussion concerning chlorine detectors and/or associated circuitry and chlorine sources at Catawba.
7. Editorially revise Selected Licensee Commitment 16.9-22 to clarify that it applies to radiation and smoke alarms only.

This activity only addresses changes to Selected Licensee Commitments 16.6-4 and 16.9-22. No changes to the remainder of the UFSAR are required as a result of this activity. UFSAR changes resulting from License Amendments 191/183 will be reported separately.

Evaluation: License Amendments 191/183 deleted chlorine detection and protection requirements contained in Technical Specification 3.7.10. Catawba submitted a license amendment request to delete these requirements since the amount of chlorine available during a single release at Catawba was within the threshold of Regulatory Guides 1.78 and 1.95, that safety related chlorine detectors and automatic control room intake isolation on a high chlorine signal should not be required. The NRC reviewed and accepted this justification and issued a Safety Evaluation stating that the Technical Specification requirements pertaining to the chlorine detectors may be deleted and that the detectors may be reclassified as non-nuclear safety related and that the automatic control room intake isolation function ias no longer required. The changes to Selected Licensee Commitments 16.6-4 and 16.9-22 were subsequently made in response to the issuance of amendments 191/183. These changes are acceptable from both a technical and licensing standpoint as indicated below:

Change 1 is acceptable because the change is editorial and removes a source for confusion concerning the meaning of the term "chlorine detection system".
Change 2 is acceptable because the new Remedial Actions better align with the function of the detectors and/or circuitry. The old references to putting flow through the HEPA filters and activated carbon adsorbers were inappropriate, since the primary function of these filters/adsorbers is not to mitigate a chlorine release accident. Extending the allowed outage time from seven to thirty days for one inoperable detector and/or circuitry in one or both intakes does not represent an unreasonable plant risk, since a redundant detector/circuitry combination is available in each intake to provide the required alarm function. Also, the likelihood of any chlorine cloud making its way to a control room intake is extremely small due to the tortuous path involved. The addition of the Selected Licensee Commitment 16.2.3 exception is acceptable because there is no reason to prohibit a plant mode or condition change as a result of an inoperable detector/circuitry. There are other Selected Licensee Commitment 16.2.3 exceptions contained in the Selected Licensee Commitment manual for various systems whose unavailability does not significantly contribute to an increase in overall plant risk, including the radiation and smoke intake alarms of Selected Licensee Commitment 16.9-22.
Change 3 is acceptable because the channel operational test and channel calibration requirements provide adequate assurance of the operability of the chorine detectors/associated circuitry.
Change 4 is acceptable because it is the addition of an editorial reference only.
Change 5 is acceptable because is supports implementation of the modification work required to downgrade the detectors to non-safety related status and to delete the automatic intake isolation function. The allowance provided by this footnote will relieve Operations from having to take the Selected Licensee Commitment Remedial Actions in response to pre-planned activities of the modification.
Change 6 is acceptable because it provides an enhanced discussion concerning chlorine sources at Catawba and the requirements for chlorine detection and protection relative to the discussion contained in Regulatory Guides 1.78 and 1.95.
Change 7 is acceptable because it is an editorial clarification stating that all chlorine detector requirements are contained in Selected Licensee Commitment 16.6-4.

This evaluation concludes that this activity does not require prior NRC approval. No Technical Specification Changes are required. UFSAR Changes will be made to Selected Licensee Commitment 16.6-4 and Selected Licensee Commitment 16.9-22.

11 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment 16.7-8, Groundwater Level

Description: Chapter 16 of the UFSAR will be revised to clarify Selected Licensee Commitment (SLC) 16.7-8 as follows:

The COMMITMENT will be revised to include a reference to a new Table 16.7-8-1 that will provide details of the operation of the groundwater monitor wells.

The REMEDIAL ACTIONS will be revised to place them in a more easily understood tabular form as opposed to the confusing manner in which it is currently stated. A new CONDITION will be added to provide direction for the allowance of one or more monitoring wells to be removed from service.

The BASES of the Selected Licensee Commitment will also be revised to include more detail on the system design and the operation and interaction of the monitoring wells and sumps.

Evaluation: The function of the Ground Water Drainage System, as stated in this Selected Licensee Commitment, is to maintain the groundwater level at or below the top of the adjacent floor slabs of the Reactor Containment Building and the Auxiliary Building. The Ground Water Drainage System is not designed to mitigate an accident nor is it deemed safety significant.

This revision will provide operators with a clearer version of the Groundwater Level Selected Licensee Commitment. The revisions will provide specific details to allow the groundwater monitoring wells to be controlled in a more consistent manner. The reliability and design function of the system is not affected by this revision.

This revision to this Selected Licensee Commitment 16.7-8 does not involve an unreviewed safety question. A change to the Technical Specifications in not required. The UFSAR will be updated as noted in this evaluation.

37 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment 16.8-5, Diesel Generator Supplemental Testing Requirements

Description: The Diesel Generators at Catawba Nuclear Station are currently required by Selected Licensee Commitment 16.8-5 to be inspected "during shutdown, in accordance with procedures prepared in conjunction with its manufacture's recommendation for this class of standby service." The commitment was imposed as part of the technical resolution to reliability concerns with the TDI engines following the crankshaft failure at Shoreham Nuclear Station in August 1983. The technical resolution to the reliability concerns was addressed in NUREG-1216. Phase I of the program "focused on the resolution of known engine component problems that had generic implications" while Phase II resulted in the "design review of a large set of important engine components to ensure their adequacy from a manufacturing standpoint, as well as operational performance". The "most critical periodic maintenance/surveillance actions" for the components identified during Phase I and II of the program were "incorporated as licensing conditions".

Operational data and inspection results were provided to the NRC for review in evaluating the relief request from the licensing conditions of NUREG-1216. This data was gathered from 1986 when the licensing conditions of NUREG-1216 were implemented until November 1992 when the Owner's Group submitted their proposal. The NRC reviewed the data and found "there is adequate justification for removing the present component-based licensing conditions". This NRC review and subsequent approved licensing relief is addressed in Safety Evaluation Report TDI-EDG-00 I -A. This resulted in the development of a Diesel Generator Maintenance Program by the members of the TDI Owner's Group with help from the engine OEM Cooper Energy Services. The current program is contained in Duke controlled document CNM-1301.00-0354. Since the development of this maintenance plan, Cooper Energy Services has dropped their 10CFR50 Appendix B Program thus creating the need for each utility to take control of their individual Diesel Generator Maintenance Programs.

The activity associated with this evaluation will address two components; 1. Performing maintenance in accordance with manufacturer's recommendations and 2. During shutdown.

1. Manufacturer's Recommendations

The Diesel Generator Manufacturer of record is Cooper Energy Services who purchased the license for the engines but did not make them. The engines were manufactured by Trans-America Deleval. The original manufacturer developed the instruction manual from which the maintenance procedures are written. Since Cooper Energy Services no longer operates under the guidance of a 10CFR50 Appendix B program, Catawba has decided to take responsibility for the Diesel Generator Maintenance Plan.

Diesel Generator maintenance at Catawba is performed per the program that was developed by the Owner's group with input from Cooper Energy Services. The source document for the plan is vendor Service Information Memo (SIM 402A). SIM402A will continue to be a source document for the Catawba diesel generator maintenance plan as well as input from the Trans-America Delaval Diesel Generator Owner's Group. The diesel generator maintenance plan at Catawba is implemented through the Duke Energy controlled document program via CNM-1301.00-0354. Therefore, revisions to the plan have to be made per minor modification with 10CFR50.59 evaluation.

The current requirements specify that the emergency diesel generators be inspected in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service. Commitment to and implementation of a reliability program that satisfies the requirements of the Station Blackout Rule and the Maintenance Rule ensures that Catawba continues to meet the respective reliability and availability goals. The program is based on monitoring, testing, and-maintenance activities. The program also uses input from industry, vendor, and Owners Group recommendations; however, it is to be bounded by the regulations and guidance of the Station Blackout and Maintenance Rules thereby eliminating the need for a redundant Selected Licensee Commitment requirement. This change is consistent with the test and inspection requirements of NUREG-1431.

The proposed activity continues to meet the design, material, and construction standards applicable to the Emergency Diesel Generators. This change will not affect the ability of the diesel generators to start, load, and provide emergency power in response to a design basis event as required per Technical Specifications. The appropriate inspections will continue to be performed to ensure the reliability and availability requirements are met as required per the Station Blackout and Maintenance Rules. The proposed change does not reduce the current reliability of the diesel generators nor does it increase the potential of increased failures. Maintenance inspections will continue to be performed per the controlled DG Maintenance Plan that includes input from the vendor through SIM 402A. The change is that Catawba will be in control of the respective DG inspections and not the OEM. As stated above, reliability and availability will continue to be tracked per the Station Blackout and Maintenance Rules.
2. During Shutdown
The current Selected Licensee Commitment requires diesel generator inspections to be performed during shutdown. This requirement does not increase the ability of a DG to perform as required per Technical Specifications. LCOs allow one DG to be removed from service for less than 72 hours without affecting power operation. Catawba is operating under the guidelines of 10CFR50.65 paragraph (a)(4) for scheduling work both innage and outage.

These guidelines are implemented and controlled per Duke Power Engineering Directives Manual EDM 201 - Engineering Responsibilities for the Maintenance Rule. These guidelines are used when removing a piece of equipment from service for maintenance activities. Catawba has been successfully removing a diesel generator from service with the plant on-line to perform required inspections that are on a frequency that is less than 18 months. The only difference will be that some of the 18 month inspections will now be performed on-line as well. Guidelines are in place to allow a diesel generator to be removed from service without impacting the health and safety to the public.

Evaluation: There are no unreviewed safety questions associated with this change. No Technical Specification changes are required. UFSAR Section 16.8-5 will be revised.

27 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment 16.9-13 - Snubbers

Description: UFSAR Chapter 16, Selected Licensee Commitment 16.9-13 provides a regulatory commitment addressing operability and inspection requirements for snubbers. This commitment was previously a Technical Specification and was approved for relocation to the UFSAR per the Safety Evaluation Report related to Amendment No. 173 to Facility Operating License NPF-35 and Amendment No. 165 to Facility Operating License NPF-52. The relocation was approved without substantive change to the former requirements.

The UFSAR change will revise the Remedial Action for Selected Licensee Commitment 16.9-13 to incorporate clarifying statements about the performance of engineering evaluations and the utilization of prior system operability evaluations. This change is needed due to complications caused by the relocation of the Technical Specification requirements to the Selected Licensee Commitment, and to address the change from a Technical Specification requirement to a Selected Licensee Commitment. The Technical Specification action previously directed that upon a condition of "one or more snubbers inoperable" a completion time of 72 hours was allowed prior to requiring the action to declare the attached system inoperable. This required action was applicable to the limiting condition for operability of Technical Specification systems. This was acceptable as a required action permitted by the Technical Specifications; however, since the snubber requirements were relocated to the UFSAR, the actions do not have the authority to delay or postpone the immediate evaluation of Technical Specification required operability and applicable actions.
Limiting conditions for operation are the lowest functional capability or performance level of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the Technical Specifications."

Therefore, this UFSAR change is revising the required actions of Selected Licensee Commitment 16.9-13 to be consistent about the determination of operability and compliance with all Technical Specification required actions upon discovery of a condition adverse to quality. Specifically, Remedial Action a) is revised to clarify that system actions are required to be entered upon the discovery of an inoperable as-found snubber and that an engineering evaluation per Selected Licensee Commitment 16.9-13.g is required. Remedial Action b) is revised to address snubbers that are rendered inoperable for testing or maintenance purposes and for which no prior system operability evaluation has been performed. Remedial Action c) is added to address snubbers that are to be rendered inoperable for testing or maintenance but for which a prior system operability evaluation has been performed.

Evaluation: The changes to Selected Licensee Commitment 16.9-13 do not alter the configuration of any structure, system, and/or component, but reinforce compliance with regulations and requirements.

These changes to Selected Licensee Commitment 16.9-13 do not change the testing requirements for snubbers. Previously, the requirements for snubbers were located in Technical Specifications and allowed inoperability of snubbers for 72 hours prior to entering required actions of the supported Technical Specification SSCs. Due to the relocation of the former requirements to the UFSAR as a commitment, the required action of Selected Licensee Commitment 16.9-13 is being revised. This revision clarifies the required actions to better ensure that Technical Specification System Actions are properly entered when appropriate. The Remedial Action is revised to a three-part format in order to better address the various scenarios that are likely to be encountered.

Remedial Action a) specifically addresses snubbers discovered to be inoperable during normal plant operation. The action requires that applicable systems and component actions be entered immediately upon such a discovery. This is consistent with the existing Technical Specification LCO requirements for inoperable components or supporting components.

Remedial Actions b) and c) address those snubbers that are rendered inoperable in order to perform testing or maintenance activities. Action b) applies to those snubbers for which no prior system operability evaluation has been performed. Since there is no basis to support continued system operability with the snubber(s) inoperable this action requires that the appropriate system and component action be entered immediately. This also is consistent with the existing Technical Specification LCO requirements for inoperable components or supporting components.

Remedial Action c) applies to snubbers that are rendered inoperable for testing or maintenance activities for which a system operability evaluation has been performed. System and component actions are not required to be entered for this situation since Technical Specification actions are only required when the system or component is found to be clearly inoperable. In this case the supported system or component has been evaluated and judged to remain operable with the snubber removed. The Remedial Action contains provisions for entering the appropriate Technical Specification actions should the conditions change such that the prior evaluation is invalidated.

There is no unreviewed safety question associated with this revision to Selected Licensee Commitment 16.9-13. No Technical Specification changes are required. UFSAR Section 16.9-13 will be revised.

13 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitment 16.9-24, "Alternate Cooling for the Chemical and Volume Control System Charging Pumps"

Description: Selected Licensee Commitment 16.9-24, "Alternate Cooling for the Chemical and Volume Control System Charging Pumps"will be revised. The commitment, surveillance and bases for this Selected Licensee Commitment are not clearly understandable.

Modifications CN-11389 (Unit 1) and CN-21389 (Unit 2) were implemented to provide a backup cooling source to support components of the Chemical and Volume Control System.s centrifugal charging pumps. This non-safety related backup cooling water comes from the Drinking Water System and supplies the Chemical and Volume Control System centrifugal charging pump motor cooler, speed reducer oil cooler, and pump bearing oil cooler. The Component Cooling System's essential supply header supplies normal cooling to these pumps. The Drinking Water System backup supply is tied into the existing the Component Cooling System supply header using a backflow preventer. On the Component Cooling System return header, drain lines are installed to route the Drinking Water System flow to the Residual Heat Removal/Containment Spray pump room sump.

The original Selected Licensee Commitment is being revised to provide additional information to more adequately define the Selected Licensee Commitment requirements and the surveillance required to assure that the commitment for supplying the alternate cooling is being met.

The Selected Licensee Commitment is provided to ensure that the Drinking Water System is available to supply both Unit 1 and 2 Chemical and Volume Control System charging pumps with cooling water upon loss of the Component Cooling System. The original COMMITMENT stated that the Drinking Water System supply would be "60 psig at the station supply header." The intent of this statement was based on a requirement of 50 psig on the outlet of the Drinking Water System pressure regulators, as determined by Engineering calculations CNC-1223.23-00-54 and CNC-1223.23-00-55. The interpretation of this COMMITMENT is not clear as to where the 60 psig requirement was to be determined. This proposed SLC revision removes the numerical requirement from the COMMITMENT and relocates it to the SURVEILLANCE section.

A Selected Licensee Commitment surveillance is being added to 16.9-24 which states: "At least once per 7 days, verify Drinking Water System pressure of 50 psig on the outlet of the station Drinking Water System pressure regulating valves." The 50 psig requirement is based on the original SLC value of 60 psig on the inlet of the regulating valves. Drawing CNM-1205.06-0410-001, pressure regulating valve design, states that the valve arrangement has a design pressure drop of 10 psid. With 60 psig on the inlet, this results in 50 psig on the valve outlets. Also, a revision to calculations CNC-1223.23-00-54 and CNC-1223.23-00-55 determined that a pressure of 40 psig on the outlet of the pressure regulating valves was sufficient to provide ample cooling water to the Chemical and Volume Control system pumps. As a conservative measure, the Selected Licensee Commitment required pressure will be established at 50 psig, as read on the discharge gauges, 0YDPG5250 and/or 0YDPG5270.

The BASES of the Selected Licensee Commitment will also be revised to provide a description of the required Drinking Water System pressure and the derivation of the committed pressure requirement.

Evaluation: The Selected Licensee Commitment function of the Drinking Water System is to supply an adequate cooling water source for the Chemical and Volume Control System charging pumps in the event that the cooling function can not be performed by the Component Cooling System.

This revision will provide a clearer version of the Drinking Water Selected Licensee Commitment. The revisions will provide specific details to allow the Drinking Water System supply pressure to be monitored in a consistent manner. The reliability and design function of the system is not affected by this revision.

The revision of this Selected Licensee Commitment does not increase the probability or consequences of accidents analyzed in the SAR. The revision provides additional information needed to interpret the required Drinking Water System function associated with the alternate cooling water supply. There is no unreviewed safety question associated with this Selected Licensee Commitment revision. No Technical Specification changes are required. UFSAR Section 16.9-24 will be revised.

41 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitments 16.8-2, 16.8-3, 16.8-4, 16.9-24, and 16.10-2

Description: Changes are being made to Selected Licensee Commitment (SLC) 16.8-2, "230KV Switchyard Systems"; SLC 16.8-3, "230KV Switchyard 125 VDC Power System"; SLC 16.8-4, "6900 Volt Shared Transformers"; SLC 16.9-24, "Alternate Cooling for Charging Pumps"; and SLC 16.10-2, "Condenser Circulating Water System". The purpose of this change is show that Work Process Manual WPM Section 607, "Maintenance Rule Assessment of Equipment Removed from Service" has been superceded by WPM 608 and WPM 609. WPM 608 covers outage risk assessments utilizing ORAM-SENTINEL. For innage risk assessment WPM 609 is used. This change replaces references to "WPM607" in these SLC's with references to "WPM 608" or "WPM 609". No technical changes were made to the SLC Manual.

Evaluation: There is no unreviewed safety question associated with this UFSAR change. No technical requirements of the SLC's were changed . The manner in which the plant and its accident mitigating equipment are designed, operated, and maintained are not affected by these changes. No accident probabilities or consequences were affected by these changes. No probabilities or consequences of equipment malfunctions were affected. No possibiliy was created for any new type of accident or equipment malfunction. No safety margins were reduced by these changes. No Technical Specification changes are required. UFSAR Sections 16.8-2, 16.8-3, 16.8-4, 16.9-24, and 16.10-2 will be revised.

90 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to Selected Licensee Commitments 16.9-7, 16.9-9 16.9-11 and 16.9-12 Bases and Selected Licensee Commitments 16.9-11 and 16.9-14 Bases

Description: The following documents are being revised: 1) Selected Licensee Commitments 16.9-7, 16.9-9 16.9-11 and 16.9-12 Bases and Selected Licensee Commitments 16.9-11 and 16.9-14 Bases. The Selected Licensee Commitment bases are revised in item (1) to reflect the applicability of each. The revisions to the Selected Licensee Commitments applicabilities were originally made with implementation of the Improved Technical Specifications. Therefore the proposed changes here are to make the bases coherent with this previously approved change (changes to Selected Licensee Commitments 16.9-7 and 16.9-9 are therefore made for consistency with a previously approved change). The bases are also revised to modify volume requirements for a cooldown to shutdown near EOC using a new limit. The new volume requirements provide additional operating margin by reasonably crediting cycle burnup and resident Reactor Coolant System boron concentration. The volume limits are transmitted to the site as part of the Core Operating Limits Report (COLR). Revision to this document is necessitated as well for Catawba Unit 1 Fuel Cycle13. Catawba Unit 2 Fuel Cycle 11 is approaching shutdown for refueling and the limits are evaluated given time in cycle with discrepancies managed through the corrective action procession.

The change is necessary because the to Selected Licensee Commitment requires the COLR stipulated volume to be treated as a floor in the Boric Acid Tank. Therefore, any increase in this volume reduces operating margin. This is particularly troublesome during boration for refueling shutdown as the higher refueling boron concentrations required either batching operations to maintain the minimum level or required the shutdown unit to use the operating unit's BAT. The proposed changes, in effect, credit the boron inserted during the cooldown to reduce the minimum volume required to attain and maintain the shutdown margin. This removes the anachronism of having sufficient volume in the BAT to borate to the required shutdown margin- even when the RCS is already borated to a concentration of or greater than the shutdown margin boron concentration.

Changes are also made to the borated water source Selected Licensee Commitment (SLC 16.9-11 and SLC 16.9-12) to reflect its applicability to the tank volumes only. The SLC surveillances reflect criteria pertaining only to the tank, as other Selected Licensee Commitments govern commitments associated with the boration flow paths and pumps. Therefore, the changes focus the SLC requirements on the boration source tanks. For Catawba these changes consist of replacing references to the Boric Acid Storage System with boric acid tank (BAT). The requirements for the Boration Systems are specified in a several SLCs covering commitments associated with flow paths, pumps and tanks. Taken together, the proposed changes described do not amend the requirements as currently stated in these SLCs.

Evaluation: A 10CFR50.59 Evaluation performed for the changes proposed in Revision 2 of calculation DCP-1552.0800-0199 applicable to Catawba Nuclear Station Unit 1 Cycle 11 addresses revisions to the bases of the Borated Water Volume (Operating and Shutdown) Selected Licensee Commitments (SLC) and each cycle's COLR. The changes made to the bases and COLRs address, in part, revisions to the minimum borated water volume requirements specified near end-of-cycle (EOC). The minimum borated water volume requirements near EOC were revised to conservatively credit cycle burnup and resident RCS boron concentration to provide boric acid tank operating margin. The minimum borated water volumes are stipulated to ensure required shutdown margin can be attained and maintained during cooldown to shutdown conditions. Values conservatively selected to bound an entire cycle unnecessarily remove operating margin from the BAT during shutdowns for refueling. Significant quantities of borated water are necessary to attain refueling shutdown boron conditions, and the unnecessarily high volume requirements represent a burden for operations and primary chemistry personnel. The revised limits continue to conservatively stipulate adequate volume to satisfy the shutdown margin requirements, while providing BAT operating margin. The minimum borated water volumes are transmitted in the Core Operating Limits Report (COLR) and therefore revisions to each cycle's COLR is necessitated. The proposed changes also revised the borated water source SLC bases discussion to reflect the SLC commitments associated with the tanks only. This change had the effect of removing the bases discussion of requirements related to the flow paths and pumps to their own respective SLCs (requirements already stipulated in these more appropriate SLCs). Therefore, the change did not remove commitments which defined system operability.

A 10CFR50.59 evaluation determined that these changes could be made without a License Amendment. No Technical Specification changes are required. USFAR Changes are required to Selected Licensee Commitment 16.9.

2 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to UFSAR Section 13.5.1.3

Description: This change modifies Selected Licensee Commitment (SLC) 16.11-5 to correct an erroneous reference to the SLC number and modifies SLC 16.13-2 by specifying the experience requirements for personnel responsible for reviewing procedures, i.e., qualified reviewers as specified in Nuclear System Directive 110. Also, it deletes the requirement that recommended changes to the Fire Protection Program be submitted to the Human Resources Manager since the organization has realigned since this was originally written.

Evaluation: These changes are administrative. No System, Structure, or Component is modified or caused to operate in a different manner from current operation. No seismic, environmental, materials, or reactivity effects are created on these Systems, Structures, or Components as a result of these changes. No new failure modes/effects or new types of system/component interactions will be introduced upon any System, Structure, or Component as a result of these changes. System, Structure, or Component behavior during steady state and transient conditions will be unaffected by any of these SLC changes. Plant response to any external phenomenon, natural or man-made, likewise will not be affected. No plant transient or accident analysis will require revision as a result of these changes. No actual testing requirements are being modified by these proposed changes; both the existing testing frequencies and testing specifications and acceptance criteria for all Systems, Structures, or Components are being maintained. No actual regulatory commitments are being eliminated or reduced by these changes.

The described changes will not increase the probability of any accident evaluated in the SAR. The changes are administrative and do not alter any technical requirements. They will have no effect upon the manner in which the plant is operated or maintained. Therefore, they will have no effect upon any accident.

There are no unreviewed safety questions associated with this UFSAR Change. No Technical Specification changes are required. UFSAR Section 13.5.1.3 will be revised.

1 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to UFSAR Section 2.3.3.3 and UFSAR Table 1-11

Description: UFSAR Table 1-11 "Regulatory Guide 1.97, Rev. 2 Review", under parameter E-17 it is currently stated that "This instrumentation was specified to have an accuracy of +/-0.1 degrees C. which is considered adequate for the intended monitoring function." The statement will be changed to state: "The sensor was specified to have a minimum accuracy of +/- 0.1 degrees C. which is considered adequate for the intended monitoring function. The instrumentation loop has an accuracy of +/- 0.22 degrees C. through and including the digital display."

UFSAR Section 2.3.3.3 will be revised to add the following sentences at the end of the section: "The ambient and delta T sensors were specified to have a minimum accuracy of +/- 0.1 degrees C. which is considered adequate for the intended monitoring function. The instrument loop has an accuracy of +/- 0.22 degrees C. through and including the digital display."

These changes clarify the position that the accuracies specified in Regulatory Guide 1.97, Revision 2, for atmospheric stability (+/- 0.15 degrees C.) are intended for the individual component and are not considered to be an instrument loop accuracy.

Evaluation: The Meteorological Monitoring System is designed to supply weather data for nuclear plant operations according to Regulatory Guide 1.23 and Regulatory Guide 1.97. Catawba Unit 1 and Unit 2 are committed to Revision 2 of Regulatory Guide 1.97. During normal operation, atmospheric stability and ambient temperature indication are used to gather data for radioactive effluent reporting requirements. From UFSAR Section 1.7.1 (Regulatory Guide 1.97 discussion) and Duke Power's response to Supplement 1 of NUREG-0737 for Catawba, Revision 4 (dated March 28, 1984): "The atmospheric stability indicating loops are Regulatory Guide 1.97 Category 3 Type E variables. Category 3 variables do not play a key role in the management of an accident but do add depth to the Category 1 and 2 instrumentation. Category 3 instrumentation is of high quality commercial grade and is selected to withstand the normal power plant service environment. The ambient temperature indicating loops are not Regulatory Guide 1.97 instruments.

There is no unreviewed safety question associated with this UFSAR change. These changes will have no effect on the probability or consequences of accidents analyzed in the UFSAR. These instruments are not accident initiators. No Technical Specification changes are required. UFSAR Section 2.3.3.3 and UFSAR Table 1-11 will be revised.

44 **Type:** UFSAR Change **Unit:** 0

Title: UFSAR Change to UFSAR Table 3-88

Description: UFSAR Table 3-88, "Design Loading Combinations for ASME Code Class 2 and 3 Components and Supports", Note 1 is being revised. The note states that the table "excludes active pumps" and states, "Refer to UFSAR Table 3-93 for loading combinations and corresponding criteria applicable for active pumps." The entire note is being deleted. The loading criteria in UFSAR Table 3-88 are the correct loading criteria for active safety related pumps. The difference in design criteria between the active safety related pumps and the inactive safety related pumps is not in the loading combinations. The actual difference in requirements is in the allowable stress and deflection criteria associated with the necessity that the active pumps must not only survive postulated events, but then pump water as their design function. Removing Note 1 from UFSAR Table 3-88 will result in the correct flow path for information. Another note in UFSAR Table 3-88 refers to UFSAR Tables 3-90 through 3-94. These tables give stress criteria and design criteria. UFSAR Section 3.9.3 gives allowable stress limits directly in the case of the Westinghouse Pump and Valve Operability Program (UFSAR Section 3.9.3.2.1) and indirectly in UFSAR Section 3.9.3.2.2, Operability Assurance of Duke Safety-Related Active Pumps and Valves, by reference through UFSAR Section 3.9.3.1.

Evaluation: The proposed change is not related to the design of components at Catawba Nuclear Station (including active pumps). The change only corrects UFSAR Table 3-88 to eliminate the incorrect statement that the loading combinations in the table exclude active pumps. The loading combinations for active pumps covered by the table are in fact the same as for other components in the table. The difference for active pumps has to do with allowable stress values. Allowable stress values are not covered in UFSAR Table 3-88.

The proposed change does not affect the design and analysis of components at Catawba Nuclear Station. The change only corrects how design information is referenced in Table 3-88 of the UFSAR. Correcting the notes in Table 3-88 of UFSAR has no relation to the probability of occurrence of an accident, and therefore does not increase the probability of occurrence of an accident previously evaluated in the SAR.

There are no unreviewed safety questions associated with this UFSAR change. No Technical Specification changes are required. UFSAR Table 3-88 will be revised.