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U. S. Nuclear Regulatory Commission
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Subject: Catawba Nuclear Station
Docket Nos. 50-413 and 50-414
1991 10 CFR 50.59 Report

Pursuant to 10 CFR 50.59, find attached a summary of Nuclear Station Modifications, Exempt Change Variation Notices, changes to the Selected License Committment Manual (SLC), procedure changes, tests, and experiments which were completed under the provisions of 10 CFR 50.59 from October 1, 1990 to September 30, 1991.

Very truly yours,

M S Tuckman

M. S. Tuckman

CRL/5059COV.492

Attachments

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Group File: 801.01
Catawba Document Control: 801.01

CATAWBA NUCLEAR STATION

Summary of Exempt Change Variation Notices
Completed Under 10CFR50.59

CE-2143 **Description:** This exempt change permanently documented the addition of the air dryer for the portable diesel air compressor.

Evaluation: The function or operability of the Instrument Air System will not be affected by the installation of the air dryer. The dryer and its components were installed in accordance with the appropriate piping and support specifications. The dryer provides compressed air at a dew point of less than -40°F which meets the required Instrument Air System dew point of less than 35°F. With the air dryer in service while using the portable diesel compressor, the supplied compressed air will be more suitable for use in the Instrument Air System.

CE-2217 **Description:** This exempt change upgrades the solenoid valves on the Feedwater Regulator and Feedwater Regulator Bypass Valves from non-QA to QA Condition 1.

Evaluation: The solenoid valves on the Feedwater Regulator Valves are purchased QA Condition 1 and did not have to be replaced. The solenoid valves on the Feedwater Regulator Bypass Valves were not QA 1 and were replaced with QA 1 ASCO solenoid valves. These new solenoid valves were mounted in the same location as the existing solenoid valves and were mounted the same as the old solenoid valves. These new solenoid valves do not create a seismic concern since no credit is taken for the seismic qualification of the Feedwater Regulator Bypass Valves. Also, the new solenoid valves will not affect the operation of the plant systems since the fit, form, and function of them is the same as the old ones.

CE-2245 **Description:** This exempt change installed a 1/16" stainless steel guard around the bottom of each doghouse water level switch.

Evaluation: The guard is bolted to the wall and does not touch the standpipe or the level switch. The guard will not interfere with water flow to the standpipe because there is a gap between the guard and the floor.

CE-2480 **Description:** This exempt change allowed for cutting or grinding portions

of the door frame angle and chipping of some concrete on doors AX632, AX635, AX635E, and AX635F to accommodate the repair of the door hinges.

Evaluation: Engineering has reviewed these changes to the door frame and surrounding concrete and has concluded that the doors will continue to perform their intended function with respect to both pressure and fire boundary requirements.

CE-2653 **Description:** This exempt change installed an improved packing configuration and associated minor changes on valve 1RN291 to enhance valve performance.

Evaluation: The modification does not affect the function or operation of the valve in any way. There is no affect to the opening/closing time of the valve, the torque requirements for the valve, nor the flow rates through the valve. In addition, all the parts associated with the modification are qualified to operate in the environment in which they are used.

CE-2677 **Description:** This exempt change replaced Moore Products 350L manual loaders with Moore Products 352B digital controllers for 1CAML0360, 0361, 0400, 0401, 0440, 0441, 0480, 0481, 0520, 0521, 0560, 0561, 0600, 0601, 0640, and 0641.

Evaluation: The manual loaders replaced are non-QA and perform no safety function. The new controllers are seismically and electrically compatible with the old controllers. To preclude any human factors concerns arising from the different appearance of the 352B faceplate from the 350L faceplate, certain items on the 352B were covered up.

CE-2678 **Description:** This exempt change replaced the Moore Products 350L manual loader with a Moore Products 352B digital controller for 1VYML0190.

Evaluation: The manual loader that was replaced is non-QA and performed no safety function. The new controller is seismically and electrically compatible with the old controller. To preclude any human factors concerns arising from the different appearance of the 352B faceplate from the 350L faceplate, certain items on the 352B were covered up.

CE-2679 **Description:** This exempt change replaced Moore Products 350L manual

loaders with Moore Products 352B digital controllers for 1NVSS5571, 5651, 5652; 1NVML1241 and 2940.

Evaluation: The manual loaders replaced are non-QA and perform no safety function. The new controllers are seismically and electrically compatible with the old controllers. To preclude any human factors concerns arising from the different appearance of the 352B faceplate from the 350L faceplate, certain items on the 352B were covered up.

CE-2758 **Description:** The purpose of this exempt change is to relocate the Conventional Low Pressure Service Water (RL) Pump C discharge valve 1RL031 downstream of the expansion joint and to install a new expansion joint.

Evaluation: The RL system does not perform any safety function and is therefore not assigned a safety class. Relocating valve 1RL031 downstream of the expansion joint will not affect the function or operability of the RL system. This arrangement is a better design because the expansion joint will be isolated from system pressure when the associated pump is isolated. The existing supports can adequately support the valve at the new location. Also, the expansion joint stainless steel stub ends will be better suited for the corrosive conditions and the thicker expansion joint liner will provide better protection for the bellows. The new stub ends and liner are designed for operation at the temperature and pressure conditions in this portion of the RL system.

CE-2762 **Description:** This exempt change revised the fully withdrawn elevation of the Rod Cluster Control Assemblies (RCCAs) for Unit 1 from 225 steps to 222 steps.

Evaluation: A complete safety evaluation was completed which addressed the following areas: RCCA Insertion Characteristics, Uncontrolled RCCA Bank Withdrawal From a Subcritical of Low Power Startup Condition, Uncontrolled RCCA Bank Withdrawal at Power, RCCA Ejection Accidents, Shutdown Margin, Reactor Internals, Peaking Factors, and Loss of Flow. No unreviewed safety questions were identified.

CE-2787 **Description:** This exempt change installed Chesterton live-loaded packing sets in the Steam Generator PORV Block Valves (1SV025, 1SV026, 1SV027 and 1SV028).

	Evaluation:	Providing live-loaded packing in the Steam Generator PORV block valves will not affect the evaluation of the SV system in the FSAR. These block valves are not required for nuclear safety.
CE-2833	**Description:**	This exempt change reduced the gain setting on the steam dump controller to eliminate Reactor Coolant System instability following "Load Rejection".
	Evaluation:	The steam dump system is not essential for safe shutdown of the unit, thus it is not designated as safety related. The steam dump system provides added flexibility in unit operation. Failure of the steam dump system will not preclude operation of any essential system. The worst operational consequences of system failure is a reactor and turbine generator trip. In the event of a complete loss of reduction in steam dump capacity, full load steam flow to the atmosphere is assured by the steam generator code safety relief valves. This change will reduce the gain of the steam dump controller to allow the steam dump valves to open and close at a slower, more conservative rate.
CE-2929	**Description:**	This exempt change modified the control circuit for 1ND002A so that the interlock with 1FW027A and 1NS043A will now be dependent on valve position only. The change also modified the control circuit for 1ND036B so that the interlock with 1FW055B and 1NS038B will now be dependent on valve position only.
	Evaluation:	Removing power from 1FW027A or 1NS043A prevents 1ND002A from opening and removing power from 1FW027A or 1NS043A prevents 1ND036B from opening. This modification will better ensure that 1ND002A and 1ND036B can be repositioned as required.
CE-3006	**Description:**	This exempt change modified RN train A to provide isolation valves and flanges in the RN pump lube injection strainer discharge piping, deleted the lube injection piping to connection 'L' and 'E' on RN pump 1A, and deleted lube injection requirements for RN pump 1A.
	Evaluation:	This modification is an enhancement to the system, does not degrade any design parameters, and cannot initiate any FSAR accident. The RN pumps are relied on during accident

scenarios so the modified portion of the system is still seismically qualified. The RN pumps function as before, are more reliable, and the modification did not create any new failure modes.

CE-3007 **Description:** This exempt change modified RN train B to provide isolation valves and flanges in the RN pump lube injection strainer discharge piping, deleted the lube injection piping to connection 'L' and 'E' on RN pump 1B, and deleted lube injection requirements for RN pump 1B.

Evaluation: This modification is an enhancement to the system, does not degrade any design parameters, and cannot initiate any FSAR accident. The RN pumps are relied on during accident scenarios so the modified portion of the system is still seismically qualified. The RN pumps function as before, are more reliable, and the modification did not create any new failure modes.

CE-3015 **Description:** This exempt change replaced the present Velocity-to-Displacement (V-D) converters for the Unit 1 Reactor Coolant Pumps. Also, an electrical band pass filter was installed to eliminate frequencies below 10 Hz and above 1000 Hz.

Evaluation: The present V-D converters are no longer manufactured. The new V-D converters are a direct replacement for the old ones. The function of the system will be unaffected by the replacement of the V-D converters. The band pass filter will eliminate spurious alarms in the Control Room due to the primary system's 5 Hz natural frequency.

CE-3075 **Description:** This exempt change permanently located backdraft dampers 1AVS-D-11 and 1AVS-D-12 downstream of their respective air flow monitors.

Evaluation: The change in weights created by moving and modifying the Annulus Ventilation (VE) duct work will not adversely affect the seismic and support requirements for this system. Moving the backdraft dampers on both trains and the air flow monitor on Train B will not cause flow rates to fall outside of the Technical Specification limits. Also, implementation of this exempt change will not reduce the negative pressure requirements or increase the annulus draw down time during

startup of the VE system.

CE-3087 **Description:** This modification changed the Auxiliary Feedwater Pump Turbine (CAPT) control valve stem from a nitrided outer surface to a chrome plate over nickel outer surface.

Evaluation: The stem base material is the same as the original stem; therefore, the strength of the stem will not be affected. Only the hardening for the surface is being changed. The weight change is negligible and no seismic qualifications are affected. Since the valve will operate as before, operation of the turbine driven CA pump is unaffected and its operability should be enhanced.

CE-3099 **Description:** This modification provides an access hole in the Main Steam (SM) system piping in order to provide access for the radiographic inspection of weld 1SM34-7 as required by ASME Code Section XI.

Evaluation: The half coupling and plug are designed to function at the temperature and pressure conditions experienced in this portion of the SM system. The half coupling and plug will maintain the integrity of the SM system and serve as a pressure boundary. In order to prevent leakage, a seal weld will be made around the plug so that the weld deposit extends fully to the outer edge of the coupling. To satisfy seismic concerns, a minimum clearance of 0.75" will be maintained between the top of the plug and the inside wall of the guard pipe. The location of the hole has been reviewed and there are no piping stress concerns.

CE-3104 **Description:** This modification removed valves 1SA014 and 1SA015 and provided threaded unions in both the leakoff lines from the Auxiliary Feedwater Pump Turbine (CAPT) trip and throttle valve (1SA145) packing area.

Evaluation: Removing these valves from the leakoff lines will provide better drainage and reduce the potential of the lines becoming clogged. Removing the valves will not adversely affect the operation of either 1SA145 or the SA (Main Steam to Auxiliary Feedwater Pump Turbine) system and will have no effect on the stress analysis. Also, the addition of the threaded unions in the leakoff lines will not affect the function of operation of these lines.

CE-3115	**Description:**	This exempt change revised the torque switch setting for valve 2NV037A.
	Evaluation:	Increasing the torque switch setting from 4 to 5 will better ensure that the valve will close. The available torque output of approximately 80 ft-lbs at a setting of 5 is well within the 200 ft-lbs design limit of the valve.
CE-3159	**Description:**	The oil seals on the component cooling water (KC) pumps have experienced premature failure. This exempt change replaced the lip type oil seals with labyrinth seals.
	Evaluation:	This change does not impact the seismic or environmental qualification of the pump. Neither does the change alter the center of gravity or mass of the pump or pump shaft significantly enough to warrant re-evaluation of the pump mounts. No controls, power supplies, or supporting structures are affected by the change. Pump operability will not be affected.
CE-3171	**Description:**	This modification replaced the epoxy bonded seat with a mechanically retained seat and installed disc travel stops on valves 1RN245 and 1RN247.
	Evaluation:	This modification does not affect the function or operation of the valve in any way. There is no affect on flow rates through the valves. In addition, all the parts associated with the modification are qualified to operate in the environment in which they are used.
CE-3172	**Description:**	This modification replaced the epoxy bonded seat with a mechanically retained seat and installed disc travel stops on valves 1RN298, 1RN305, and 1RN307.
	Evaluation:	This modification does not affect the function or operation of the valve in any way. There is no affect on flow rates through the valves. In addition, all the parts associated with the modification are qualified to operate in the environment in which they are used.
CE-3175	**Description:**	This modification deleted the snubbers from supports 1-A-KC-4108 and 1-A-KC-4177.
	Evaluation:	Neither the function or operability of the KC system will be

affected by deleting the snubbers from the supports listed above. Engineering has performed the appropriate calculations and analyses relating to the reduction of snubbers to ensure the affected Component Cooling (KC) system piping will be adequately supported during normal operation and seismic events.

CE-3179 **Description:** This change adds additional data for verification of torque switch settings and adds recalculated setpoint data to the "Torque Switch Setting Sheets" in compliance with Duke's response to Generic Letter 89-10 for the following valves: 1BB147B, 148B, 149B, 150B; 1CA015A, 018B, 085B, 116A; 1FW033A, 49B; 1KCC37A, C40B, 345A, 364B, 394A, 413B, 429B, 430A; 1NC054A; 1ND001B, 002A, 028A, 036B, 037A, 058B, 059B, 090, 091; 1NI009A, 010B, 103A, 118A, 121A, 135B, 136B, 147B, 150B, 152B, 153A, 154B, 162A, 332A, 333B, 334B, 438A; 1NM039, 040, 207A, 210A; 1NV015B, 037A, 044A, 055A, 066A, 077A, 089A, 091B, 188A, 189B, 202B, 203A, 312A, 314B, 477, 865A, 876, 877; 1WL450A, and 451B.

Evaluation: This change will not affect any present signals that initiate valve motion. Valve operator speed and capacity are unaffected. Open and closure times of these valves are unchanged. This modification does not affect the valves ability to perform their design function during an accident. No safety related functions are added to or deleted from these valves.

CE-3194 **Description:** This modification provides flush connections for the Nuclear Service Water (RN) Train A Pump Motor upper bearing oil coolers and replaces the RN Train A oil cooler throttle valves with 1/2" diaphragm valves which will be less susceptible to fouling and will provide better flow control.

Evaluation: This modification is an enhancement to the system and does not degrade any design parameters. The RN pumps are relied on during accident scenarios so the modified portion of the system is still seismically qualified. The RN pumps will function as before, and no new failure modes were identified.

CE-3195 **Description:** This modification provides flush connections for the Nuclear Service Water (RN) Train B Pump Motor upper bearing oil coolers and replaces the RN Train B oil cooler throttle valves

with 1/2" diaphragm valves which will be less susceptible to fouling and will provide better flow control.

Evaluation: This modification is an enhancement to the system and does not degrade any design parameters. The RN pumps are relied on during accident scenarios so the modified portion of the system is still seismically qualified. The RN pumps will function as before, and no new failure modes were identified.

CE-3199 **Description:** This exempt change updates the Unit 2 Steam Generator (S/G) tube sheet maps to include all tubes plugged to date for each Unit 2 S/G. This evaluation provides justification for tube plugging activities.

Evaluation: The largest number of tubes plugged in any Unit 2 S/G is 23 out of a total of 4674 tubes which is well below the 10% allowed by the ECCS analysis model. The installation of tube plugs will not degrade primary boundary integrity, but help to maintain it. The worst case scenario due to failure of a tube plug is judged to be a single tube rupture event. A single tube rupture is a design basis accident and is addressed in the Catawba FSAR.

CE-3205 **Description:** This modification replaced the refueling cavity seal with a differently designed seal.

Evaluation: The entire assembly was tested for compatibility with borated water similar to that present during the seals design use. The new design reduces the possibility of seal failure because it does not require internal air pressure and its construction provides an additional impact barrier. Analysis for environmental compatibility, leakage under pressure, seal splice leakage, seismic response, and impact from a dropped fuel assembly indicate the seal will function an designed.

CE-3213 **Description:** This modification provides an access hole in the Main Steam (SM) system piping in order to provide access for the radiographic inspection of weld 1SM-1A-F as required by ASME Code Section XI.

Evaluation: The half coupling and plug are designed to function at the temperature and pressure conditions experienced in this portion of the SM system. The half coupling and plug will maintain the integrity of the SM system and serve as a

pressure boundary. In order to prevent leakage, a seal weld will be made around the plug so that the weld deposit extends fully to the outer edge of the coupling. To satisfy seismic concerns, a minimum clearance of 0.75" will be maintained between the top of the plug and the inside wall of the guard pipe. The location of the hole has been reviewed and there are no piping stress concerns.

CE-3226 **Description:** This modification revised design documents and installed the correct NAMCO model limit switches for the following: 1CFLL0334, 0424, 0515, 0605, and 1NVLL4660.

Evaluation: The fit, form, and function of the replacement limit switches is identical to the existing switches. In addition, the replacement limit switches were EQ approved by the engineering department.

CE-3249 **Description:** This modification provided electrical connections for Auxiliary Feedwater Low Suction Pressure response time testing.

Evaluation: This modification wired a set of spare cable conductors between 1EATC5 and 1AFWTCP to allow testing of the B Train circuitry in 1AFWTCP. The spare cables used are Safety Related QA-1. The connections to the field devices are controlled by the maintenance procedure for testing the affected circuitry.

CE-3256 **Description:** This modification added a 1.0 Microcurie source to each of the steamline radiation monitors (1EMF26, 1EMF27, 1EMF28, and 1EMF29).

Evaluation: The addition of the keep-alive source to the PC boards of each of the detectors will create a high enough level of radioactivity to avoid the monitors sensing a loss of signal. The sources themselves are of a low enough level of radioactivity that they will not create a significant increase in radioactivity in the near vicinity of the monitors.

CE-3257 **Description:** This modification added a 1.0 Microcurie source to each of the steamline radiation monitors (2EMF10, 2EMF11, 2EMF12, and 2EMF13).

Evaluation: The addition of the keep-alive source to the PC boards of

each of the detectors will create a high enough level of radioactivity to avoid the monitors sensing a loss of signal. The sources themselves are of a low enough level of radioactivity that they will not create a significant increase in radioactivity in the near vicinity of the monitors.

CE-3270 **Description:** This exempt change revised the vendor maintenance manual CNM-1205.00-1889 for NAMCO limit switches to reflect additions concerning model number agreement with switch operation.

Evaluation: The fit, form, and function of the limit switches will change only in the fact that the correct model number will be used for actuation and that the switches will not have to be adapted for operation. This change will also prevent negation for the Equipment Qualification of the limit switches.

CE-3282 **Description:** This modification provided manual throttling capability for valves NI173A and NI178B.

Evaluation: Seismic qualification is not degraded for any piece of equipment since the actual changes in weight, center of mass, moments, and other relevant characteristics are insignificant. This modification introduces no new modes which could initiate any accident previously evaluated in the FSAR. Modification of the control circuits will not degrade the safety function of the valves. The new operating characteristics will be incorporated into procedures where appropriate.

CE-3289 **Description:** This modification replaced valve 1NI102 with Valve Item No. CSR-021.

Evaluation: Neither the function or operation of the NI system will be affected by replacing valve 1NI102 with Valve Item No. CSR-021. The design of the replacement valve is very similar to the existing valve and the new valve will function identically to the old valve. Piping and support modifications are not required for installation of the new valve and there are no seismic concerns.

CE-3290 **Description:** This modification replaced valve 1ND031 with Valve Item No. CSR-035.

Evaluation: Neither the function or operation of the ND system will be

affected by replacing valve 1ND031 with Valve Item No. CSR-035. The design of the replacement valve is very similar to the existing valve and the new valve will function identically to the old valve. Piping and support modifications are not required for installation of the new valve and there are no seismic concerns.

CE-3310 **Description:** This modification updated documents to allow the mirror insulation and steel supports to be removed and blanket insulation to be installed on the Pressurizer.

Evaluation: Neither the function nor the operability of the Pressurizer will be affected by replacing the mirror insulation with blanket insulation. Duke Specification CNM 1206.13-00-0001, Rev. 1, allows the replacement of mirror insulation with blanket insulation inside Containment. The new blanket insulation and supports for the Pressurizer meet all the conditions addressed in the Duke Specification.

CE-3312 **Description:** This modification capped the Unit 1 Incore Instrument Thimble J-07 at the seal table due to thimble damage as a result of the thimble being bent.

Evaluation: As a conservative measure to protect against a small break LOCA, thimble J-07 was capped with a QA-1 instrument cap. Technical Specifications require 75% (44 of 58) of the thimbles to be used during Incore Flux Mapping activities. This modification will bring the total number of capped thimbles to 2 for Unit 1. Therefore, there are still 56 of 58 thimbles available for flux mapping.

CE-3325 **Description:** This exempt change modified valves 1NI56, 67, 78, and 90 by replacing the disk with a soft seat disk assembly.

Evaluation: Neither the function or the operability of the NI system will be affected by modifying the disk assembly in the above mentioned NI valves. The soft seat disk assembly has been tested in this type of air operated valve and has proven to be acceptable for this application. The soft seat disk assembly is acceptable for use in the NI system up to temperatures of 300 degrees F and will not degrade due to radiation affects.

CE-3329 **Description:** This exempt change verified and documented the change in the item number for valve 1ND008 from 9J-602 to 9J-550.

	Evaluation:	Neither the function or the operability of the ND system will be affected by modifying valve 1ND008 to Valve Item No. 9J-550. The new valve is very similar to the existing valve and will function identically to the old valve.
CE-3334	**Description:**	This modification replaced relief valve 1RN290 with Valve Item No. CSR-047.
	Evaluation:	Neither the function or operation of the RN system will be affected by replacing valve 1RN290 with Valve Item No. CSR-047. The design of the replacement valve is very similar to the existing valve and the new valve will function identically to the old valve. Piping and support modifications are not required for installation of the new valve and there are no seismic concerns.
CE-3372	**Description:**	This modification replaced the Personnel Airlock latch assemblies with latch assemblies that have better thermal protection.
	Evaluation:	The new latch assemblies will function exactly the same as the old latch assemblies with some minor wiring changes to the latches respective terminal boxes.
CE-3379	**Description:**	This exempt change revised the CNS Maintenance Coating Schedule to allow both the vendor coating and the application of coatings per specification 103-I as non-service level I coatings on the Reactor Vessel internals lifting rig.
	Evaluation:	The surface area of the coatings on the reactor vessel internals lifting rig was already accounted for as non-service level I coatings. Evaluation indicates that the loss of this coating will not adversely affect any plant system's, specifically the containment spray system's recirculation pump screen assembly.
CE-3384	**Description:**	This modification restored the outer seal ring of the Unit 1 Equipment Hatch Barrel to its original configuration. This is required since the outer seal ring was damaged during outage related work.
	Evaluation:	The restoration was accomplished by mechanical means (grinding) and did not involve any thermal process. In addition, the grinding will not remove enough material to

violate code minimum wall thickness. This modification did not affect the sealing capabilities of the Equipment Hatch.

CE-3389 **Description:** This modification replaced the valve spring packs for valves 1NC033A and 1NC035B with a different model spring pack.

Evaluation: The new spring pack model is QA-1. Discussions between Engineering and the valve manufacturer regarding this replacement verified that the new spring packs will have the same fit, form, and function.

CE-3403 **Description:** This modification replaced the high pressure fitting at the seal table and repositioned the Unit 1 incore instrument thimble B-6.

Evaluation: This modification will not reduce the number of thimbles required by the Technical Specifications during incore flux mapping activities. No significant affects will be made on the flux mapping capability for Unit 1.

CE-3404 **Description:** This modification added a note to CN-1499-NC11.02 in regards to resolution of PIR 1-C91-0223.

Evaluation: ~~This is~~ a change to a drawing to reflect the as built configuration ~~and does not change~~ plant operation in any way.

CE-3405 **Description:** This modification repaired two vertical tears that were discovered in basket 9-9 of ice condenser bay 5.

Evaluation: The ice condenser is not used for any phase of normal plant operations. It has no interface with any plant system used either for power generation of shutdown cooling. The ability of the ice condenser to perform its accident mitigation functions is not degraded by this modification. The seismic qualification of the ice condenser is not degraded. No other plant system used for accident mitigation is affected by the modification. The margin in ice mass relative to the safety analysis ia not degraded by the modification.

CE-3406 **Description:** The Main Steam Safety Valves have a specific bearing band for the spring washer. The existing allowable band ranges from 1/8" minimum to 3/16" maximum while the actual band maximum in 1/4". This exempt change increased the spring

washer bearing band maximum to 1/4".

Evaluation: Neither the function nor the operability of the SV or SM system will be affected by increasing the spring washer bearing band maximum. The valve manufacturer and the Engineering Department have reviewed this modification and determined that the function of the main steam safety valves will be unaffected.

CE-3412 **Description:** This change provides the Hydroset 1566 testing device constant of 0.352 applicable to the main steam safety valves.

Evaluation: Neither the function nor the operability of the SV or SM system will be affected by increasing the spring washer bearing band maximum. The valve manufacturer and the Engineering Department have reviewed this modification and determined that the function of the main steam safety valves will be unaffected.

CE-3418 **Description:** This modification replaced relief valve 1NI086 with Valve Item No. CSR-016.

Evaluation: Neither the function or operation of the NI system will be affected by replacing valve 1NI086 with Valve Item No. CSR-016. The design of the replacement valve is very similar to the existing valve and the new valve will function identically to the old valve. Piping and support modifications are not required for installation of the new valve and there are no seismic concerns.

CE-3427 **Description:** This modification installed 0.109" orifices in port "P" of the fast acting solenoid valves for Main Turbine CVs 1,2,3,4 and CIVs 1,2,3,4,5 and 6 to prevent low ETS pressure.

Evaluation: A General Electric Change Notice identified this problem and recommended that 0.109" orifices be installed in the fast acting solenoid valves "P" port to prevent low ETS pressure during valve testing. This change has been evaluated by GE and determined to be acceptable.

CE-3438 **Description:** This exempt change updated general arrangement drawings and electrical equipment drawings to reserve an area in the Auxiliary Building for snubber test equipment. This exempt change also set up and issued a vendor O/M manual for the

snubber test equipment.

Evaluation: Seismic interaction concerns, additional HVAC loading concerns, floor loading concerns, electrical noise interactions, fire concerns, and internally generated missile concerns were all evaluated and found to be acceptable.

CE-3446 **Description:** This modification redistributed the gains for the Unit 1 OTΔT setpoint by replacing some of the resistors in the PCS cards.

Evaluation: The redistribution of the OTΔT gains will not increase the probability of an inadvertent reactor trip. Given the nature of the hardware changes, there are no adverse effects in the loads on the PCS power supplies. Compliance with seismic, environmental, and Appendix R criteria is not adversely affected. Neither fission product barriers nor the source term evaluation is affected.

CE-3447 **Description:** This modification redistributed the gains for the Unit 2 OTΔT setpoint by replacing some of the resistors in the PCS cards.

Evaluation: The redistribution of the OTΔT gains will not increase the probability of an inadvertent reactor trip. Given the nature of the hardware changes, there are no adverse effects in the loads on the PCS power supplies. Compliance with seismic, environmental, and Appendix R criteria is not adversely affected. Neither fission product barriers nor the source term evaluation is affected.

CE-3455 **Description:** This modification raised 1EMF33s refrigerated dryer approximately 6 inches higher than its present elevation. The inlet and outlet piping was also modified to use tubing with compression type fittings.

Evaluation: Neither flowrates nor sample integrity will be degraded since the new piping/tubing layout is less constrictive and uses welded or compression type joints. The inputs and outputs for 1EMF33 were not changed, and, therefore, the function of the system remains the same.

CE-3456 **Description:** This modification raised 2EMF33s refrigerated dryer approximately 6 inches higher than its present elevation. The inlet and outlet piping was also modified to use tubing with compression type fittings.

Evaluation: Neither flow rates nor sample integrity will be degraded since the new piping/tubing layout is less constrictive and uses welded or compression type joints. The inputs and outputs for 1EMF33 were not changed, and, therefore, the function of the system remains the same.

CATAWBA NUCLEAR STATION

Summary of Nuclear Station Modifications
Completed Under 10CFR50.59

Note: The first number designates the unit on which the modification was performed (i.e. CN-2#### means the modification was performed on unit 2, a 1 means unit 1 and a 5 means it is shared by both units).

CN-10058 **Description:** This NSM essentially replaced the carbon steel RN piping on the supply and discharge sides of the VG (Diesel Generator Starting Air System) Aftercoolers with 2" stainless steel pipe.

Evaluation: Reliability of the aftercoolers was improved based on less flow restrictions due to crud buildup in the pipes. No equipment has been degraded and no common mode failures were introduced.

CN-10067 **Description:** This modification provided a positive displacement pump to dewater the Post Accident Liquid Sampling panel sump.

Evaluation: Pipe interaction and stress analysis calculations were reviewed and found to be acceptable. This modification will not adversely affect the operation of the panel. The Class E portion of the system is separated from the Class B portion by Class B manual isolation valves and a check valve.

CN-10318 **Description:** This modification replaced carbon steel piping with stainless steel piping in order to alleviate fouling and corrosion problems around RN Radiation Monitors 1EMF45A and 45B.

Evaluation: The environmental, seismic and Appendix R qualifications of the affected part of the RN system are unchanged. The modified portion of the RN system performs a monitoring function and will essentially function as before.

CN-10588 **Description:** This modification relocated blowdown flow control valves 1BB24, 65, 69, and 73 to allow straight discharge into the blowdown tank.

Evaluation: No safety system is degraded by this modification and no functional change was made to the system. An Appendix R review was conducted.

CN-10893 **Description:** This modification provided floor drains needed to collect spills from valve packing and flanges that leak potentially contaminated water on elevation 543.

Evaluation: There will be no appreciable load increase for the floor if the drain clogs up and fills the curbing with water. No safety related equipment is in the area of the curbing so flooding of safety related equipment inside the curbing is not a concern. No concerns with seismic interaction or flooding of equipment on elevation 522 due to pipe break exists.

CN-10925 **Description:** This modification provided the Boric Acid Tank with a means of automatic venting.

Evaluation: This modification improved the availability of the Boric Acid Tank by providing automatic venting capability and overpressure relief. The tank will function essentially as before.

CN-10942 **Description:** This modification modified selected valve operators in the NS, ND, CA, CF, KC, RN, SM, and SV systems to resolve concerns for these valves not attaining their desired positions due to insufficient torque switch settings relative to the resistance through the stroke travel.

Evaluation: The torque bypass contacts are being adjusted to a travel span of 50% ± 25%. These valves will operate identically to the way they presently operate but should be more reliable in attaining their desired positions when required to change. No new failure modes are created as a result of the torque bypass travel span increase and a premature motor trip resulting from a high torque is reduced.

CN-10977 **Description:** This modification replaced globe valve 1NV475 with a double disc gate valve.

Evaluation: This valve is an isolation valve used for maintenance only. The globe valve was not used as a pressure reducing valve or for throttling purposes; therefore, a gate valve is an adequate substitute. The valve coefficient for the new gate valve is acceptable. The new valve will be the same material, class, and have the same design conditions as before.

CN-11005 **Description:** This modification removes or replaces snubbers identified as

no longer needed.

	Evaluation:	The piping systems and support/restraints reanalyzed for snubber reduction were done so in consideration of loads and load combinations and allowable stress criteria as prescribed in the Catawba FSAR, and in accordance with appropriate procedures for QA-1 and QA-4 calculations. No redundancy or separation criteria is violated by this modification.
CN-11067	**Description:**	This modification placed a manual bypass for P-14 safety signal (Steam Generator High-High Level) to prevent feedwater isolations during Modes 4, 5 and 6.
	Evaluation:	A seismic analysis has been performed for the main control board and found to be satisfactory. An appropriate Appendix R review was satisfactorily conducted.
CN-11107	**Description:**	This modification adds a recirculation loop to the Boric Acid Tank (BAT).
	Evaluation:	The BAT recirculation loop is qualified for normal environmental and vibrational conditions. The qualification of neither the NV, WL, nor the NB system is degraded as a result of this modification. Neither any fission product barriers nor any source term evaluation is adversely affected by the modification. The ability to maintain the BAT boric acid concentration within required limits is not adversely affected by the modification.
CN-11119	**Description:**	This modification added seal-in circuitry to the Auxiliary Feedwater Pump Turbine (CAPT) overspeed trip lights on the main control board and the SSF.
	Evaluation:	Addition of the seal-in circuitry will not affect the trip function, only the individual trip lights. The relays that were added are environmentally qualified and will be seismically mounted by standard procedure.
CN-11123	**Description:**	This modification replaced the stuffingbox on the Unit 1 RN pumps.
	Evaluation:	The new stuffingbox will perform all the functions of the old stuffingbox and the additional taps will only be in use during maintenance periods. The operation of the RN pumps will

not be adversely affected by this modification. No new failure modes are introduced as a result of this modification.

CN-11146	**Description:**	This modification redesigned the present method for flushing the RN to CA FWP piping.
	Evaluation:	The KC, RN, and CA system as well as the Auxiliary Shutdown Panel will function as before. The potential for effluent to be flushed from the KC system to the RN system has been evaluated and determined acceptable.
CN-11163	**Description:**	This modification replaces the turbine-generator (TGN) Electro-Hydraulic Control (EHC) with a digital turbine control system which uses state of the art technology.
	Evaluation:	The turbine trip functions listed in the FSAR are preserved, and the anticipated frequency of a turbine trip is not increased. The reliability of the new EHC system is at least the same as that of the current EHC. The safety analysis of turbine overspeed and turbine trip are unaffected by the modification and remain bounding.
CN-11168	**Description:**	This modification installed the Digital Feedwater Control System (DFCS) on Unit 1.
	Evaluation:	Input signal validation is employed and no failure mechanisms were identified that would adversely affect safety. The systems purpose and the manner of implementing this purpose remain unchanged as a result of this modification and all failures of this system lead to conditions that are analyzed specifically or are in the FSAR. All signals to and from this system are isolated from safety related equipment.
CN-11169	**Description:**	This modification revised existing supports on Feedwater system piping in the Turbine Building.
	Evaluation:	The function of the feedwater system is not altered by these revisions to the supports. The probability of any Feedwater System accident or any other accident is not increased since no system including the feedwater system is adversely affected by the revision to the supports.
CN-11176	**Description:**	This modification replace the Diesel Generator (D/G) Fuel Oil (FD) and Lube Oil (LD) rigid corner piping with

flexhoses.

Evaluation: The flexhose segments and accessories to be installed have been evaluated in an acceptable substitution report. The design pressures and temperatures of the flexhose selected match or exceed the design pressure and temperatures of the other parts of the FD and LD Systems. The stresses and deflections corresponding to a combination of normal loads and seismic forcing functions have been found to be within acceptable limits.

CN-11178 **Description:** This modification replaced the existing Rochester Instruments sequence of events recorder (SER) with the Dranetz Model 22 SER. A new printer and stand were also added.

Evaluation: The function of the SER is not altered, and no new failure modes are introduced. The addition of the new printer and printer stand have been shown to not adversely affect surrounding equipment and are not required to be mounted as QA 4.

CN-11182 **Description:** This modification adds support brackets to guide non-essential thermocouple cables into the reactor vessel instrument ports.

Evaluation: This modification does not affect the function of any operating system. The support brackets for the thermocouple cables will reduce the likelihood of failures due to bending of connections. The new supports were evaluated for potential stress problems and found acceptable.

CN-11183 **Description:** This modification performs HED related improvements on the Main Control Boards and the Auxiliary Feedwater Pump Turbine Control Panel.

Evaluation: The changes performed in this modification are cosmetic in nature and will not affect the performance of any system. This modification has no adverse impact on the seismic qualification of the affected main control boards. No control circuits will be changed as a result of this modification.

CN-11188 **Description:** This modification upgrades the software for the Inadequate Core Cooling Monitor-86 (ICCM-86) to improve the operation of the ICCM system based on a Westinghouse Field Change Notice.

	Evaluation:	The ICCM performs no function during normal plant operation. The interfaces it has with equipment used during normal plant operations will not be affected by this modification. The ICCM functions are unchanged as a result of the modification. The seismic and Appendix R qualification of the ICCM are not affected by this modification.
CN-11191	**Description:**	This modification installed ultrasonic level instrumentation on the Reactor Coolant (NC) system piping on both the B and the C hot legs.
	Evaluation:	Material compatibility was considered in the design of this modification. Seismic and stress evaluations were performed for this modification. The impact on the integrity of the NC system piping was reviewed. An Appendix R review was conducted.
CN-11192	**Description:**	This modification changed the orientation of the Feedwater Control Bypass Valve (FCBV) and its internal configuration.
	Evaluation:	This modification was examined for all considerations involving flow, failure modes, accidents and installation standards and has been found acceptable.
CN-11195	**Description:**	The Diesel Generator Engine Jacket Water Cooler Inlet Valves, RN232A and RN292B, were replaced with a more reliable and more easily maintainable butterfly valve.
	Evaluation:	The valves will function as before with respect to signals and operating times. Although the C_v is less, the flow capability is adequate. The operator is also adequately sized. Materials, stress analysis, and pressure/temperature parameters have been considered.
CN-11201	**Description:**	This modification installed a set a accelerometers on each of the four Lower Containment Ventilation Units (LCVUs).
	Evaluation:	This modification will have no adverse impact on equipment used for normal plant operations. No new failure modes were identified. No accident mitigation system or equipment is affected by this modification. The ability of the LVCUs to maintain lower compartment temperature within the limits of safety analysis is not degraded by the modification. No

fission product barrier is affected. Furthermore, no source term analysis is affected.

CN-11216	**Description:**	This modification replaced gate valves BB8, 10, 19, 21, 56, 57, 60, and 61 with gate valves with a different Internal configuration and a slightly larger Electric Motor Operator (EMO).
	Evaluation:	No current function of these valves is being added to, deleted from or altered. The new valves operate in the same manner and are of the same class and construction material as the old valve.
CN-11222	**Description:**	This modification replaced the control room analog processor Ice Condenser (NF) temperature chart recorder with a chart recorder that has a digital processor.
	Evaluation:	The NF temperature recorder is provided to trend NF temperature and to alarm on high temperature in the ice bed. It has neither a control function nor a control interface with any system used for normal plant operation. The reliability of the ice bed temperature monitoring will not be impaired by this modification.
CN-11227	**Description:**	This modification changed the tubing downstream of the discharge side of the existing vent valves in the NW Normal and Assured Makeup Water lines to piping.
	Evaluation:	The material specifications, design limits, and safety classification of the drain lines through the isolation valves will be the same as the Assured Makeup Water Lines. The modifications do not affect the integrity or the availability of either the RN or the NW System to respond to an accident.
CN-11239	**Description:**	This modification will relocate a 3/4" line from upstream of NI95A to just downstream of the same valve. Check valve NI471 will be installed in the new line.
	Evaluation:	Materials to be added under this modification are compatible with the existing design. The valves will still function as designed as containment isolation valves.
CN-11241	**Description:**	The Fuel Handling Area Ventilation (VF) controls will be modified to prevent shutdown of the VF Exhaust Fan upon

indication of high radiation levels in the unit vent stack or upon detection of smoke upstream of the VF Exhaust Filter Units.

Evaluation: Removal of the trips on high radiation level in the unit vent stack and on smoke upstream of filters will result in the increase of the availability of the VF exhaust fans to perform their design function. Neither of these trips are safety related and no credit is taken for them in the safety analysis. Compliance with the appropriate seismic, Appendix R, and separation criteria is unchanged. No changes are made in any of the interfaces between VF and any system used for any phase of plant operation.

CN-11242 **Description:** This modification changed control circuit wiring and associated documentation for 1CA2, 4, and 6; 1FW01A and 32B; 1ND32A and 65B, 1RF389B, 447B and 457B; and 12 KC valves in accordance with the guidance in Generic Letter 89-10.

Evaluation: All the valves will be rewired to include open-torque bypass switches. Valve operator speed and capacity are unaffected. Open and closure times are unaffected. Any signals which would result in valve motion are unaffected.

CN-11243 **Description:** This modification installed drain lines downstream of 1RN236, 296, and 2RN236.

Evaluation: The design of this modification will not functionally affect any system during any mode of operation. The design will prevent excessive leakage and potential line breaks.

CN-11250 **Description:** This modification added a branch pipe, isolation valve, weld flange and blind flange to the Unit 1 Nuclear Service Water supply crossover line to facilitate temporary insertion of a seal plug.

Evaluation: The additional mass and change to the center of gravity of the RN pipe was analyzed and found to be acceptable for pipe restraints and stress analysis concerns. The modification used materials and methods equivalent to those already in the RN system.

CN-11254 **Description:** This modification machined a 2.7 inch diameter penetration

in the secondary side shell of steam generator 1B and provided bolted closure.

Evaluation: The closure meets all ASME Section III Code requirements and is consistent with the design parameters associated with the Steam Generator. Stress analysis demonstrates structural integrity of the shell penetration and closure for all conditions in compliance with ASME Section III code requirements. The previously analyzed steam line break accident bounds any potential failure of the shell closure assembly.

CN-11259 **Description:** This modification performed a leak repair on the Steam Generator secondary side manway.

Evaluation: The repair met all ASME Section III code requirements and was consistent with the design pressures and temperatures associated with the Steam Generator. The structural integrity of the Steam Generator is not compromised by this modification.

CN-20073 **Description:** This modification placed a manual bypass for P-14 safety signal (Steam Generator High-High Level) to prevent feedwater isolations during Modes 4, 5 and 6.

Evaluation: A seismic analysis has been performed for the main control board and found to be satisfactory. An appropriate Appendix R review was satisfactorily conducted.

CN-20279 **Description:** This modification provided floor drains needed to collect spills from valve packing and flanges that leak potentially contaminated water on elevation 543.

Evaluation: There will be no appreciable load increase for the floor if the drain clogs up and fills the curbing with water. No safety related equipment is in the area of the curbing so flooding of safety related equipment inside the curbing is not a concern. No concerns with seismic interaction or flooding of equipment on elevation 522 due to pipe break exists.

CN-20396 **Description:** This modification removes or replaces snubbers identified as no longer needed.

Evaluation: The piping systems and support/restraints reanalyzed for snubber reduction were done so in consideration of loads and

load combinations and allowable stress criteria as prescribed in the Catawba FSAR, and in accordance with appropriate procedures for QA-1 and QA-4 calculations. No redundancy or separation criteria is violated by this modification.

CN-20558 **Description:** This modification provided more uniform flow from each steam generator (S/G) sample line to EMF34 by installing a Linear Kinetic Cell (LKC), a manually controlled throttle valve, and a flow meter in each sample line.

Evaluation: The flow valves and instrumentation will be installed in accordance with applicable QA standards and the equipment is rated for the design temperatures, pressures, and flow rates for this part of the system, and no operating or control function will be added or deleted.

CN-11216 **Description:** This modification replaced gate valves BB8, 19, 56, and 60 with gate valves with a different Internal configuration and a slightly larger Electric Motor Operator (EMO).

Evaluation: The function of these valves and its method of implementation remains unchanged. No current function of these valves is being added to, deleted from or altered. The new valves operate in the same manner and are of the same class and construction material as the old valve.

CN-20572 **Description:** This modification modified and relocated the flow restricting orifice installed in the main feedline (loops C and D) downstream of the feedwater bypass piping takeoff and moved check valve 2CF168 downstream and increased the line size downstream of 2CF96.

Evaluation: The orifices are properly designed and function exactly as before. Structural mechanics, stress analysis and hanger design have been reviewed and found acceptable.

CN-20576 **Description:** This modification changed the orientation of the Feedwater Control Bypass Valve (FCBV) and its internal configuration.

Evaluation: This modification was examined for all considerations involving flow, failure modes, accidents and installation standards and has been found acceptable.

CN-20582 **Description:** This modification allowed testing of certain valves in order

to update document CNM-1205.00-1997 (Torque Switch settings for Rotork and Limitorque EMO's).

Evaluation: The operation of these valves does not constitute excessive wear to the valves nor a radiological hazard to test personnel. There will be no changes to the valves after completion of the testing, and no control functions will be altered.

CN-20615 **Description:** This modification relocated the Upper Surge Tank (UST) level instrumentation high point connection from the riser between the UST and the Upper Surge Dome Tank (USDT) to the USDT to the USDT vent line to the condenser.

Evaluation: The ability to monitor UST level is not degraded by the modification. The availability of the UST as a preferred CA source is not adversely affected by the modification.

CN-20631 **Description:** This modification performed a leak seal repair on valves 2SV25B and 2SV27A.

Evaluation: The ability of the valves to function as designed is not adversely impacted by this modification. No new failure modes were identified. The operability of the valve was verified after completion of leak repair.

CN-20632 **Description:** The Fuel Handling Area Ventilation (VF) controls will be modified to prevent shutdown of the VF Exhaust Fan upon indication of high radiation levels in the unit vent stack or upon detection of smoke upstream of the VF Exhaust Filter Units.

Evaluation: Removal of the trips on high radiation level in the unit vent stack and on smoke upstream of filters will result in the increase of the availability of the VF exhaust fans to perform their design function. Neither of these trips are safety related and no credit is taken for them in the safety analysis. Compliance with the appropriate seismic, Appendix R, and separation criteria is unchanged. No changes are made in any of the interfaces between VF and any system used for any phase of plant operation.

CN-20637 **Description:** This modification provided the option for on-line leak sealing repair of valve 2CA61 by installing bonnet studs with injection washers below the bonnet nut.

	Evaluation:	No new failure modes were identified, and the valve operation will not be affected. The valve is still Class B and seismically designed. The sealant to be used has been categorized for this use by the Power Chemistry section.
CN-50308	**Description:**	This modification provided additional security barriers for certain areas at the protected area boundary and vital area boundaries.
	Evaluation:	Design considerations were made concerning drainage, HVAC flow, and attachment to seismic walls and found to be satisfactory.
CN-50375	**Description:**	This modification added an oil reclaim line to both YC Control Area Chillers, CRA-C-1 and CRA-C-2.
	Evaluation:	The modification only involves connecting copper tubing between the evaporator and condenser of the chillers. There will be no impact on containment leakage or fuel cladding integrity. The installation of the copper tubing has been seismically analyzed and the tubing has been reviewed for piping interactions. The modification has no impact on the chillers' design basis requirements of maintaining proper temperature control in the Control Area.
CN-50385	**Description:**	This modification will increase the corrosion protection provided to the Lube Oil Tanks, Fuel Oil Tanks and lines.
	Evaluation:	The cathodic protection system is not nuclear safety related. The function of the D/G's to operate post accident is not affected by this modification. The modification was designed to the proper QA Conditions and industry standards.
CN-50414	**Description:**	This modification provided the option of routing the service water strainer backwash to the RN pump pit.
	Evaluation:	A stress analysis and interaction review was completed satisfactorily. The RN strainers, pumps, and pump pit equipment are not degraded and no new failure modes were found. The RN pump suction flow and temperature are not adversely affected and the operation of the RN System equipment and heat sinks are not affected.
CN-50415	**Description:**	This modification built a cofferdam and settlement pond near

the SNSWP to retain silt and waste water which accumulate during dredging around the SNSWP intake structure.

Evaluation: The effect of a cofferdam failure on the SNSWP was evaluated. The results show that the SNSWP intake would not be inundated upon complete failure of the cofferdam. The integrity and availability of the SNSWP following a seismic event will not be degraded by the modification.

CATAWBA NUCLEAR STATION

Summary of Changes to the SLC Manual
Completed Under 10CFR50.59

SLC 16.2.7	**Description:**	This change reduced the maximum allowable surveillance extension from 50% to 25% in 16.2.7.
	Evaluation:	This change brings the requirements of the SLC manual in line with the requirements of Technical Specification 4.0.2. This change improves the chances of discovering unreliable equipment and is more conservative than the current requirements.
SLC 16.9.5	**Description:**	This change provided a complete list of fire barriers and sealing devices to SLC 16.9.5.
	Evaluation:	This change will prevent further PIRs/LERs due to missed fire watches.
SLC 16.9.6	**Description:**	This change deleted fire zones 128 and 130 from Table 16.9-3.
	Evaluation:	This change was editorial in nature in that it corrected a mistake made in the original SLC.

CATAWBA NUCLEAR STATION

Summary of Procedure Changes Completed
Under 50.59 From 10/1/90 - 9/31/91

TN/1/B/2488/CE/01A, Initial Issue

Implementation Procedure TN/1/B/2488/CE/01A provides guidance for the modification of the control and alarm circuitry for Radiation Monitor (EMF) 1EMF31. The changes will incorporate alarm delays during normal backflush cycles and pump restart delays following backflushes to allow the pump time to stop its backward rotation before restarting. In order to make these changes, some relays will have to be added or changed and some wiring modifications performed. In order to perform these electrical changes, the Turbine Building Sump (WP) Sample Pump and flow switches 1WPFS5100 and 5120 will need to be isolated. To accomplish the isolations, a breaker for the pump will need to be isolated. To accomplish the isolations, a breaker for the pump will need to be opened and two fuses pulled to isolate the flow switches. These isolations will disable the sample pump, radiation monitor 1EMF31, flow switches 1WPFS5100 and 5120, the pump indicating lights on 1TBOX0057, the flow alarm light on 1EMF31, and the low flow annunciator for 1EMF31 in the control room. Since 1EMF31 will be out of service, the Tech. Spec. requirements for manual sampling will be followed during implementation of this Exempt Change. The equipment and sampling system affected by this Exempt Change are not safety related or required to bring the unit to a safe shutdown. Only the WP radiation monitoring system and equipment are affected.

Based on the above discussion, it is concluded that there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1195/00/01A, Initial Issue

NSM CN-11195, Rev. 0, replaces the BIF butterfly valves 1RN232A and 1RN292B, which are currently installed in Diesel Generator Rooms 1A and 1B. These valves serve as the Diesel Generator Engine Jacket Water Cooler (KD HXs) inlet isolation valves. Posi-Seal valves will be installed at these locations. This procedure provides guidelines for the replacement of valve 1RN232A.

Valve 1RN232A is interlocked to open when Diesel Generator (D/G) 1A starts and supply cooling water (RN) to the Diesel Generator engine jacket water cooler. The valve closes when the Diese stops. Therefore, replacing valve 1RN232A requires Diesel Generator 1A to be taken out of service. The redundant diesel, Diesel Generator 1B, is not affected by this procedure and will be available to supply emergency power. If a Unit's diesel is out of service or down for maintenance, then the shared valves normally powered from that channel are provided with manual switchover to the other Unit's diesel of corresponding channel. However, due to isolations, the corresponding Unit 2 Diesel will likewise be out of service during the replacement of valve 1RN232A. This procecure requires that the valve replacement be performed under the constraints of Tech. Spec. 3.8.1.1 which requires that D/G 1A be returned to service within 72 hours. This ensures that the required D/G's are available to supply emergency power to all accident mitigation systems. Due to the redundancy in components, the loss of one train will not prevent the accident mitigation systems from performing their intended functions and achieve safe shut down.

No system affected by this procedure initiates accidents. The loss of one Diesel Generator KD Heat Exchanger has already been evaluated.

Based on the above considerations, no unreviewed safety questions are determined to be associated with the implementation of this procedure.

TN/1/A/1195/00/02A, Initial Issue

NSM CN-11195, Rev. 0, replaces the BIF butterfly valves 1RN232A and 1RN292B, which are currently installed in Diesel Generator Rooms 1A and 1B. These valves serve as the Diesel Generator Engine Jacket Water Cooler (KD HXs) inlet isolation valves. Posi-Seal valves will be installed at these locations. This procedure provides guidelines for the replacement of valve 1RN292B.

Valve 1RN292B is interlocked to open when Diesel Generator (D/G) 1B starts and supply cooling water (RN) to the Diesel Generator engine jacket water cooler. The valve closes when the Diesel stops. Therefore, replacing valve 1RN292B requires Diesel Generator 1B to be taken out of service. The redundant diesel, Diesel Generator 1A, is not affected by this procedure and will be available to supply emergency power. If a Unit's diesel is out of service or down for maintenance, then the shared valves normally powered from that channel are provided with manual switchover to the other Unit's diesel of corresponding channel. However, due to isolations, the corresponding Unit 2 Diesel will likewise be out of service during the replacement of valve 1RN292B. This procedure requires that the valve replacement be performed under the constraints of Tech. Spec. 3.8.1.1 which requires that D/G 1B be returned to service within 72 hours. This ensures that the required D/G's are available to supply emergency power to all accident mitigation systems. Due to the redundancy in components, the loss of one train will not prevent the accident mitigation systems from performing their intended functions and achieve safe shut down.

No system affected by this procedure initiates accidents. The loss of one Diesel Generator KD Heat Exchanger has already been evaluated.

Based on the above considerations, no unreviewed safety questions are determined to be associated with the implementation of this procedure.

PT/0/A/4400/08 Retype, Changes 0 to 57 Incorporated

The purpose of this procedure is to ensure that all safety related components receive adequate cooling water during a faulted Engineered Safety Features (ESF) situation from the Nuclear Service (RN) System. This procedure does not create an unreviewed safety question.

There are two different balances performed using this procedure. One is the full flow balance, which includes all necessary plant components such as the Component Cooling Heat Exchangers, the Containment Spray (NS) Heat Exchangers, and the Diesel Generator Engine Cooling Water (KD) Heat Exchangers. The other flow balance will only set flows pertaining to the pump itself, such as bearing lubrication, motor cooling, and bearing oil cooling.

During the full flow balance, the trains are first isolated from each other. Neither train is made inoperable, although the component under which maintenance was performed is considered to be inoperable. The component flows

are adjusted and throttle valves locked once all flows are within the specified range.

Upon the cleaning of a NS Heat Exchanger, the associated RN train will be considered operable; only the isolated heat exchanger will be considered inoperable. During the post-cleaning flow balance, the addressed heat exchanger will gradually be brought back into service while the key flows are monitored for their maintained operability.

During the pumphouse flow balance, only the pump under test is placed in service. The crossover valves are not closed, and pump discharge pressure is lowered to the pressure obtained during the previous full flow balance by use of inservice components and additional NS Heat Exchangers. Again, neither train of RN is made inoperable by the procedure. Operators are fully aware of the status of the NS Heat Exchangers at all times.

The RN pumps are not operated outside design conditions.

PT/1/A/4600/03A, Change #56

This change consisted of correcting an incorrect page number, adding a sign-off space which was removed by mistake from the last retype, and revising the allowed maximum flow rates of both Reactor Makeup Water Pumps when the Boron Dilution Mitigation System is inoperable in Modes 3, 4, or 5.

These new flow rates, along with the Source Range Neutron Flux Monitors, ensure that adequate time is available for the operator to recognize and terminate a dilution event prior to a loss of shutdown margin.

The new flow rates were obtained from a Duke reanalysis, dated May 2, 1989, of the Boron Dilution Accident. The reanalysis resulted from a Westinghouse bulletin dated July 19, 1988, concerning potential non-conservatisms in the existing boron dilution analysis. This change is being incorporated to remove the non-conservatisms from the Tech. Spec. surveillance, although the Tech. Specs. have not been changed at this time.

This procedure change does not create an unreviewed safety question.

PT/2/A/4600/03A, Change #41

This change revised the allowed maximum flow rates of both Reactor Makeup Water Pumps when the Boron Dilution Mitigation System is inoperable in Modes 3, 4, or 5.

These new flow rates, along with the Source Range Neutron Flux Monitors, ensure that adequate time is available for the operator to recognize and terminate a dilution event prior to a loss of shutdown margin.

The new flow rates were obtained from a Duke reanalysis, dated May 2, 1989, of the Boron Dilution Accident. The reanalysis resulted from a Westinghouse bulletin dated July 19, 1988, concerning potential non-conservatisms in the existing boron dilution analysis. This change is being incorporated to remove

the non-conservatisms from the Tech. Spec. surveillance, although the Tech. Specs. have not been changed at this time.

This procedure change does not create an unreviewed safety question.

OP/1/A/67C0/01, Change #191

This change incorporates new Power Range (P/R) Nuclear Instrumentation System (N S) Calibration data obtained per PT/1/A/4600/05A. This change replaces page 1 of Table 2.2 with new data.

OP/1/A/6700/01, Unit One Data Book, Table 2.2 is used to record the 100% Full Power Calibration Currents (at Axial offsets of +20%, 0%, and -20%) and the M factors for each of the Power Range Excore Detectors. Data is obtained for this table only by approved procedure PT/1/A/4600/05A, "Incore/Excore Calibration", or PT/1/A/4600/05D, "Interim Incore/Excore Calibration". The data recorded on this table is used by Instrument and Electrical to adjust the Axial Flux Difference (AFD) calculating circuitry and Operator Aid Computer (OAC) programs. It may also be used to manually calculate AFD and Quadrant Power Tilt Ratio (QPTR) if the OAC is inoperable.

Since AFD is used to dynamically adjust both the Overtemperature Differential Temperature and Overpower Differential Temperature setpoints, the data herein is safety related.

This change does not create an unreviewed safety question.

OP/2/A/6700/01, Change #121

This change updated curve 1.2.1, Temporary Rod Withdrawal Limits. Temporary Rod Control Limits need to be established per PT/0/A/4150/21, "Temporary Rod Withdrawal Limits Determination" performed during Catawba Unit 2 Cycle 4 (C2C4) Zero Power Physics Testing.

This curve is intended to supply the Operators with a reference to the results of PT/0/A/4150/21, "Temporary Rod Withdrawal Limits Determination" performed on 9/13/90 for C2C4. This curve is required by Tech. Spec. 3.1.1.3.a to ensure that the Moderator Temperature Coefficient (MTC) is negative at 100% power (and less negative than required by Tech. Spec. 3.1 at other power levels.)

The results provide for Temporary Rod Withdrawal Limits at 0% power and between 85% and 100% power. At all other power levels the MTC will be within Tech. Specs.

Data from CNNE 1553.05-08 (C2C4 Startup and Operational Report) Figure 9 indicates that the predicted Hot Full Power (HFP) Boron Concentration for 4 Effective Full Power Days (EFPD) is 1243 ppmB. Since this is less than the most limiting boron concentration, Temporary Rod Withdrawal Limits will no longer be necessary beyond 4 EFPD.

This curve does not affect plant systems, structures, or components in any way. The procedure calculating this curve uses Design data and data from physics testing. This change does not create an unreviewed safety question.

PT/2/A/4250/03E Retype, Changes 0 to 9 Incorporated

This procedure retype verifies proper flow balance of Auxiliary Feedwater (CA) lines to the Steam Generators (S/Gs) and verifies the proper stroking of CA check valves as required by Tech. Specs. and ASME Section XI, respectively. This test is performed in Mode 3 prior to going into Mode 2. During all portions of the test, all three CA pumps and associated flow paths are maintained in an operable condition. Modifications installed to fail open the CA flow control valves do not defeat the fail open capability of the CA flow control valves on a valid Auto Start Signal. Reclosure of the flow control valve can be immediately established by closure of the test switch, and only one out of the three CA pumps is tested at any given time.

This test injects water from the CA pumps to the S/Gs, which are near their normal operating pressure. During the injection period, normal feedwater flow to the affected S/G is isolated. This is a design function of the CA system up to approximately 3% Reactor Power. S/G high level (Feedwater Isolation) and S/G low level (CA Auto Start) capability is maintained at all times. Adequate margin above the Safety Injection Setpoint on low steam line pressure (725 Psig) is maintained by providing test termination criteria of 900 Psig.

The CA system is used in its normal standby readiness alignment as specified by the FSAR. No Unreviewed Safety Questions exist by the performance of this procedure.

PT/2/A/4200/09, Change #63

This evaluation applies to change number 63 of PT/2/A/4200/09, Engineered Safety Features (ESF) Actuation Test. This change adds additional load to the actual load for Diesel Generator (D/G) 2A for equipment that was not operated during the A train Blackout section of the ESF test. These components have already been retested to verify they will start on a blackout signal. Values for loads are obtained from FSAR table 8.3.1-1. The change also deletes 2VP7A from the enclosures where response times are recorded. 2VP7A exceeded its allowable stroke time; however, Tech. Spec. 3.6.1.9 requires that the valve be sealed closed during modes 1 through 4. As a result, power is removed from this valve with the valve in its safety position (closed) during the modes in which it is required to be able to close in a specified time.

No unreviewed safety question is created by this change.

PT/2/A/4200/09, Change #57

This procedure change was written to remove power from Containment Spray (NS) valves 2NS12B and 2NS15B during the performance of section 12.4 and to restore power after the test. This will ensure that no water will be sprayed into upper containment during section 12.4. The load for these valves is not

needed in section 12.4, and the valves will be timed in section 12.5. Another part of this change will rack NS pump 2B to test for section 12.5. This will keep NS 2B pump from starting in order to ensure that no water will be sprayed into upper containment during performance of section 12.5. The breaker for NS pump 2B will be returned to the "as found" position by the current procedure. Unit 2 is required to be in Mode 5 by this procedure, and NS is not required to be operable in mode 5. Thus, an unreviewed safety question is not created by this change.

PT/2/A/4200/09C, Change #1

This change will allow the testing of the Boron Dilution Mitigation System (BDMS) in a mode where a charging pump is operating to provide seal injection flow to the Reactor Coolant pump seals. The original procedure was designed to be performed when all the charging pumps are off. This change will cause a swap of suction sources from the Volume Control Tank (VCT) to the Refueling Water Storage Tank (FWST) while the charging pump is running. This change of suction sources with the pump operating is the original design of the system.

This system is designed to supply high boron concentration water from the FWST in case an unexpected dilution is taking place. The test prerequisites verify that a planned boron dilution is not in progress. Tech. Specs. 3.1.2.5 and 3.1.2.6 ensure that the FWST boron concentration is maintained at a minimum of 2000 ppmB in Modes 1 to 6. In any case, the injection of borated water from the FWST will only cause the shutdown margin of the Reactor Coolant system to increase.

During the entire test, both trains of BDMS will remain operable and capable of performing their intended function. No unreviewed safety question is created by performance of this procedure.

PT/1/A/4200/34, Change #15

Change #15 to PT/1/A/4200/34 revises the Valve Inservice Test (IWV) stroke time value for Containment Purge (VP) valves to 10 seconds, to eliminate the need to maintain the valves with an unnecessary limiting value of 5 seconds. The required action for stroke time increases is also changed. Trending will be evaluated on a case-by-case basis. The evaluation will be performed by the responsible engineer.

The VP system is only operated in mode 5 or lower. Timing the valves to close within 10 seconds will ensure containment closure for high radiation and high relative humidity. The valves will continue to be tested on a cold shutdown frequency. No unreviewed safety question is created by this procedure.

PT/2/A/4250/02E, Change #34

This change provides the necessary steps to verify the proper operation of the S/G Hi-Hi Level (P-14) Blocking switch. Signals will be generated in the process cabinets, and proper blocking and unblocking functions of the switch will be verified. Since the Feedwater Isolation function from P-14 is only

required for Modes 2 and 1, and this test can only be performed in Modes 5, 6 and No Mode, there is no impact on Tech. Specs.

The addition of these steps in the procedure does not change the final actuation of the P-14 circuit. If the P-14 fails to block the initiating signal, the equipment affected does not change from the original intent of the procedure. Therefore no new accident scenarios are created, and the probability of an evaluated accident scenario is not increased. There is also no new equipment malfunction created since the addition of the P-14 block circuit has already been evaluated in the modification package.

No unreviewed safety question exists.

MP/0/A/7300/03, Change #1

This is a previously approved and fully reviewed procedure to which a new section is being added, section 11.27. Section 11.27 is titled "Pneumatic Control Valve Removal, Replacement, and Corrective Maintenance". The new section provides a method of procedural documentation and guidance for removal, replacement, and corrective maintenance of the dryer pneumatic control valves. Incorporation of the dryer pneumatic control valves into this procedure is to consolidate inspections and activities under one procedure for this component. Enclosure 13.1 has been upgraded to document requirements of the new section 11.27.

This addition verifies or restores the pneumatic control valves back to their original material and operating condition. It does NOT create an unreviewed safety question.

MP/0/A/7450/34, Change #1

The following changes were made to this procedure:

* Step 4.1.1 was changed to provide further clarification.

* Changed the word Condenser to Compressor in step 11.2.5.

* Placed statement (Crack open valve 5) at the beginning of step 11.5.1.5.

* Deleted original steps 11.6.5 through 11.6.13 and added the following new steps:

  11.6.5 Visually check liquid level on site glass, (located on the cooler shell).
  NOTE: Do not exceed 2 1/2 bolt level on the cooler site glass. Storage tanks may contain reserve refrigerant.
  11.6.6 When Freon level reaches 2 1/2 bolts on the cooler site glass, close liquid line valve 7.
  11.6.7 Turn off pumpout Compressor.
  11.6.8 Close all valves (1 thru 11).

The above changes were made to provide further guidance and clarification to the maintenance personnel performing the work under the procedure.

Tech. Spec. 3/4.7.6 (two trains of YC operable) is affected by this procedure. Operations has the responsibility and the procedures for compliance with this Tech. Spec. Maintenance will be performed on these chillers when Tech. Specs. allow, per Operations procedures.

Maintenance performed under this procedure has been reviewed against approved vendor manuals, design documents, and station procedures to ensure that corrective maintenance controlled by this procedure will return this chiller to as-built / as-designed condition. These actions will ensure the chiller's compliance with FSAR accident analysis. Since the chiller will be returned to as-designed conditions, the possibility, consequences or probability of a malfunction will be reduced. Therefore, no unreviewed safety question exists.

OP/1/A/6200/12, Change #25

This change consists of the following items:

1) Adding a step which, in conjunction with Chemistry's procedure, will provide administrative controls to ensure the Reactor Make-up Water Storage Tank (RMWST) makeup flow will not exceed its overflow capability. The RMWST has an overflow capacity of 40 to 60 gpm, depending on the level in the overflow discharge tank (Recycle Holdup Tank -- RHT). Each of the two Make Demineralizer Vacuum Deaerator Discharge pumps, which are used for RMWST makeup, can provide a maximum of 40 gpm to the RMWST. By only allowing one pump to be in operation during the makeup, overflow capacity will not be exceeded.

2) Revising the allowed maximum flow rates of both Reactor Makeup Water Pumps when the Boron Dilution Mitigation System (BDMS) is inoperable in Modes 3, 4, or 5.

These new flow rates, along with the Source Range Neutron Flux Monitors, ensure that adequate time is available for the operator to recognize and terminate a dilution event prior to a loss of shutdown margin.

The new flow rates were obtained from a Duke reanalysis, dated May 2, 1989, of the Boron Dilution Accident. The reanalysis results from a Westinghouse bulletin, dated July 19, 1988, concerning potential non-conservatisms in the existing boron dilution analysis. This change is being incorporated to remove the non-conservatisms from the Tech. Spec. surveillance.

OP/2/A/6200/12, Change #12

This change consists of the following items:

1) Adding a step which, in conjunction with Chemistry's procedure, will provide administrative controls to ensure the RMWST makeup flow will not exceed its overflow capability. The RMWST has an overflow capacity of 40 to 60 gpm, depending on the level in the overflow discharge tank (RHT).

Each of the two Make Demineralizer Vacuum Deaerator Discharge pumps, which are used for RMWST makeup, can provide a maximum of 40 gpm to the RMWST. By only allowing one pump to be in operation during the makeup, overflow capacity will not be exceeded.

2) Revising the allowed maximum flow rates of both Reactor Makeup Water Pumps when BDMS is inoperable in Modes 3, 4, or 5.

These new flow rates, along with the Source Range Neutron Flux Monitors, ensure that adequate time is available for the operator to recognize and terminate a dilution event prior to a loss of shutdown margin.

The new flow rates were obtained from a Duke reanalysis, dated May 2, 1989, of the Boron Dilution Accident. The reanalysis results from a Westinghouse bulletin, dated July 19, 1988, concerning potential non-conservatisms in the existing boron dilution analysis. This change is being incorporated to remove the non-conservatisms from the Tech. Spec. surveillance.

PT/2/A/4200/09, Change #60

This change to the Engineered Safety Features (ESF) Actuation Periodic Test procedure specifies retests of components (2KC56A, 2NI9A, 2NI10B, and 2NV91B) for which no response times were obtained during the ESF test. 2NI9A and 2NI10B had power removed during the test to prevent Reactor Coolant System volume from increasing. 2KC56A and 2NV91B went to their safety position during the ESF test; however, no response times were obtained from the Response Time Testing program on the Operator Aid Computer (OAC).

The retests for 1KC56A and 1NV91B are essentially identical to the Valve Inservice (IWV) tests performed during cold shutdown, with the exception of the initiating signal. Instead of using the pushbutton in the control room, a jumper will be placed to stroke the valves to their closed position. During the test of 2NI9A and 2NI10B, pressurizer level may increase slightly, and there may be a slight temperature transient. Prior to opening the valves, pressurizer level will be verified to be at a low level. Normal charging will be isolated. Letdown may also be increased. The valves will only be open for a short period of time. Each valve has a stroke time of approximately 5 seconds, and the valves may be closed as soon as they reach their open position. Nothing is being done to any of the valves being tested to hinder their performance. The margin of safety as defined in the bases to Tech. Specs. is not reduced by this change to the procedure. No Unreviewed Safety Question is created by this procedure.

TN/5/A/0078/00/01A, Initial Issue

NSM CN-50078, Rev. 0, will improve the reliability of the Control Area Ventilation and Chilled Water (VC/YC) systems, simplify maintenance, and reduce radiation exposure. This procedure will provide guidelines to add individual "ON-AUTO" switches of various Train A VC/YC equipment, interlock Train A Control Room return air damper with its respective Smoke Purge fan, provide a Train A "sequencer signal control circuit" switch on the main

control board, delete Train A VC/YC temperature switches and associated annunciators, and downgrade Train A Chilled Water Pump differential pressure switch from safety to non-safety.

No work will begin on this procedure until the critical Train A VC/YC isolation dampers and switchgear Air Handling Units (AHUs) are operational for retest purposes. Also, work will not begin on this A Train VC/YC procedure until B Train is in operation and determined that this train can pressurize the Control Room adequately to satisfy Tech. Spec. 4.7.6.e.3. This procedure will be implemented under Tech. Spec. 3.7.6. This Tech. Spec. allows one Train of VC/YC to be out of service for 7 days. Train A VC/YC will be out of service during the implementation of this procedure.

No other system will be prevented from performing any function important to safety while this work is being performed. All equipment affected by this procedure and the design intent of this modification will be completely tested by Performance procedure TT/0/A/9100/56.

The margin of safety defined in the bases of the Technical Specifications is unaffected. An unreviewed safety question does not exist.

TN/5/A/0078/00/02A, Initial Issue

NSM CN-50078, Rev. 0, will improve the reliability of the VC/YC systems, simplify maintenance, and reduce radiation exposure. This procedure will provide guidelines to add individual "ON-AUTO" switches of various Train B VC/YC equipment, interlock Train B Control Room return air damper with its respective Smoke Purge fan, provide a Train B "sequencer signal control circuit" switch on the main control board, delete Train B VC/YC temperature switches and associated annunciators, and downgrade Train B Chilled Water Pump differential pressure switch from safety to non-safety.

No work will begin on this procedure until the critical Train B VC/YC isolation dampers and switchgear AHUs are operational for retest purposes. Also, work will not begin on this B Train VC/YC procedure until A Train is in operation and determined that this train can pressurize the Control Room adequately to satisfy Tech. Spec. 4.7.6.e.3. This procedure will be implemented under Tech. Spec. 3.7.6. This Tech. Spec. allows one Train of VC/YC to be out of service for 7 days. Train B VC/YC will be out of service during the implementation of this procedure.

No other system will be prevented from performing any function important to safety while this work is being performed. All equipment affected by this procedure and the design intent of this modification will be completely tested by Performance procedure TT/0/A/9100/56.

The margin of safety defined in the bases of the Technical Specifications is unaffected. An unreviewed safety question does not exist.

TN/5/A/0078/00/05A, Initial Issue

NSM CN-50078, Rev. 0, will improve the reliability of the VC/YC systems, simplify maintenance, and reduce radiation exposure. This procedure will provide guidelines to install a hand-operated butterfly valve in duct between discharge of smoke purge fan 1CR-SPF-1 and the unit vent and to install a shaft seal on fan 1CR-SPF-1.

No work will begin on this A Train VC/YC procedure until B Train is in operation. This procedure will be implemented under Tech. Spec. 3.7.6. This Tech. Spec. allows one Train of VC/YC to be out of service for 7 days. Train A VC/YC will be out of service during the implementation of this procedure.

No other system will be prevented from performing any function important to safety while this work is being performed. Before returning the smoke purge fan 1CR-SPF-1 to service, Construction/Maintenance Department (CMD) personnel will assure the valve operates freely and properly, and Performance will perform a flow balance of Train A VC/YC per PT/0/A/4450/08.

The margin of safety defined in the bases of the Technical Specifications is unaffected. An unreviewed safety question does not exist.

TN/5/A/0078/00/06A, Initial Issue

NSM CN-50078, Rev. 0, will improve the reliability of the VC/YC systems, simplify maintenance, and reduce radiation exposure. This procedure will provide guidelines to install a hand-operated butterfly valve in duct between discharge of smoke purge fan 2CR-SPF-1 and the unit vent and to install a shaft seal on fan 2CR-SPF-1.

No work will begin on this B Train VC/YC procedure until A Train is in operation. This procedure will be implemented under Tech. Spec. 3.7.6. This Tech. Spec. allows one Train of VC/YC to be out of service for 7 days. Train B VC/YC will be out of service during the implementation of this procedure.

No other system will be prevented from performing any function important to safety while this work is being performed. Before returning the smoke purge fan 2CR-SPF-1 to service, CMD personnel will assure the valve operates freely and properly, and Performance will perform a flow balance of Train B VC/YC per PT/0/A/4450/08.

The margin of safety defined in the bases of the Technical Specifications is unaffected. An unreviewed safety question does not exist.

TN/1/A/0925/00/01A, Initial Issue

The Boric Acid Tank (BAT) does not have adequate overpressure protection. This problem could result in failure of the tank. The purpose of NSM CN-10925, Rev. 0, is to modify the below the diaphragm vent line so automatic venting capability is provided. The purpose of this procedure is to provide guidance for the modifications to the below the diaphragm vent line.

Implementing this procedure will require isolation and draining of the BAT. The Operations Group will coordinate the isolations necessary to implement this procedure. The modification to the vent line may be performed during an outage and in Modes 5, 6, and No Mode. The BAT will be out of service during the modification. The systems and equipment affected by this procedure can be out of service during Modes 5, 6, and No Mode.

Testing for NSM CN-10925, Rev. 0, will be performed in accordance with the Post Modification Testing program at the station. The new relief valve will be tested and verified to relieve at the design setpoint. This testing will be performed before and after the valve is installed in the system. The test of the valve before installation will verify the valve has been properly designed by the manufacturer. The test of the valve after installation will verify the new test connection will perform its intended function, which is to provide a flowpath for in-line testing of the relief valve. The testing after installation of the relief valve will be performed in Modes 5, 6, or No Mode while the tank is out of service. In addition, the BAT level protection channels will be calibrated to verify proper operation of the instrumentation. Since the BAT is designed as an atmospheric tank, pressure testing of the new piping and components is not required. All of the testing referenced in this evaluation is identified on the Post Modification Testing Plan and has been reviewed by the system expert to ensure the testing identified adequately addresses the concerns of the post modification testing program.

The Operations Group will control the isolations required to implement this procedure. Post modification testing will be performed to ensure the relief valve performs its intended function.

Based on this discussion, there are <u>no</u> unreviewed safety questions associated with the implementation of this procedure.

OP/2/A/6700/01, Change #123

OP/2/A/6700/01 (Unit Two Data Book) Table 2.2 is a table of data for use by plant personnel.

<u>FULL POWER CURRENTS</u>

This section is used to record the 100% Full Power currents for +20, 0, and -20% axial offset and M factors for each of the Power Range Excore Detectors. Data is obtained for this section only by use of approved procedures such as PT/2/A/4600/05E, Refueling ENB Calibration, or PT/2/A/4600/05A, Incore/Excore Calibration. Other tests may supply data to this section but in all cases the tests must be approved tests.

The data recorded here is used by Instrument and Electrical (IAE) personnel to adjust the Axial Flux Difference (AFD) calculating circuitry and Operator Aid Computer (OAC) programs. It may also be used to manually calculate AFD if the OAC is inoperable.

Since AFD is used to dynamically adjust both the Overtemperature Differential Temperature and Overpower Differential Temperature setpoints, the data here is safety related.

This change replaces page 1 of Table 2.2 to reflect new Power Range Currents from the interim incore/excore calibration performed at 30% power per PT/2/A/4600/05D.

Information in OP/2/A/6700/01 (Unit Two Data Book) is changed only by approved procedure change. No equipment other than the Nuclear Instrumentation System (NIS) is affected by Table 2.2. The safety margin will not be decreased.

PT/2/A/4400/06A, Change #7

Step 8.9 in the procedure is changed to ensure that the Refueling Water Storage Tank (FWST) heater setpoints are adjusted to maintain an FWST temperature of 93 degrees. This temperature will ensure that a sufficient temperature differential exists between the FWST and Nuclear Service Water (RN) to perform a heat capacity test. The setpoints will be returned to normal upon the completion of the test.

Tech. Specs. specify that the FWST temperature must remain between 70 and 100 degrees. The temperature is required to be maintained above 70 degrees to preclude possible damage to the containment vessel due to the inadvertent operation of the containment spray system. The temperature will still be maintained within the allowable limits as specified by Technical Specifications. The margin of safety as defined in the bases to Tech. Specs. will not be reduced by this change.

No unreviewed safety question is created by this change.

MP/0/B/7650/124, Initial Issue

This procedure is to be used as a guideline for station engineers and design engineers for vibration testing of station components, and as documentation that testing has occurred on station components. This procedure may be used in the absence of a work request for data collection. However, if any work is to be performed in conjunction with this testing, a work request will be required, and appropriate corrective maintenance procedures will be employed.

This procedure includes a checklist similar to the nuclear safety evaluation checklist which must be completed and reviewed by a qualified reviewer prior to any testing. In addition to the checklist, the test plan must be written and approved by the Maintenance Engineering Services (MES) engineer responsible for the component, as well as responsible Operations or Chemistry personnel.

This procedure is to be used for diagnostic testing only. No changes to plant components or systems may be performed under this procedure. No unreviewed safety question exists as a result of this procedure.

MP/0/A/7200/01 Retype, Changes 0 to 4 Incorporated

This procedure has been revised; the changes to the procedure are as follows:

Section 11.0 Step 11.3.3 was added to inspect the washers for flatness and smooth finish.

Section 11.0 Step 11.5.8 was revised to include more detail about the correct sequence for replacing the packing in the valve bonnet.

Section 11.0 Step 11.6 was revised to include more detail on the correct method for setting the valve stem travel.

Enclosure 13.1 Was revised to add appropriate sign-offs.

Tech. Spec. 3/4.7.1 may be affected by this procedure. Operations has the responsibility and the procedures for compliance with this Tech. Spec. Maintenance will be performed on the governor valve when Tech. Specs. allow, per Operation's procedures. This revision will clarify and assure that maintenance activities will return the governor valve to as-designed conditions.

The changes made by this revision are to incorporate lessons learned from performing maintenance on the valve. Design documents and station procedures have been reviewed to ensure that the corrective maintenance controlled by this procedure will return the governor valve to as-built/as-designed condition. These actions will ensure the governor valve's compliance with FSAR accident analysis. Since the governor valve will be returned to as-designed conditions, the possibility, consequences, or probability of a malfunction will be reduced. Therefore, no unreviewed safety question exists.

MP/0/A/7450/26 Change #2

This safety evaluation is for Change #2 to MP/0/A/7450/26. The following changes have been made to the procedure:

- Added steps 11.2.6.1 through 11.2.6.4 to allow removal of fan sheeve and bushing from shaft.

- Added steps 11.2.10.1 through 11.2.10.10 to allow removal of pillow block bearing assembly.

- Deleted steps 11.4.3 through 11.4.16.

- Added new steps 11.4.3 through 11.4.6 to allow reinstallation of drive end pillow block bearing, sheeve, and allow belt alignment and tensioning.

- Sign-offs were added to Enclosure 13.1 (Data Sheets) to document torque valves for steps 11.4.3.4 and 11.4.5.1.

The changes being made to this procedure have been reviewed against approved vendor manuals, design documents, and station procedures to ensure that the corrective maintenance controlled by this procedure will return the fan to

as-built/as-designed conditions. These actions will ensure the fan's compliance with FSAR accident analysis. Since the fans will be returned to as-designed conditions, the possibility, consequences, or probability of a malfunction will be reduced. Therefore, no unreviewed safety question exists.

OP/0/B/6250/09, Retype #7, Changes 0 to 22 Incorporated

This retype is a major rewrite and contains numerous changes which include the following:

1) Former enclosures dealing with precoating Condensate Polishing Demineralizers (CPD) and the use of Solution A were deleted. This information was incorporated into Enclosures 4.1 and 4.2.

2) The maximum flowrate through a CPD was changed from 5450 GPM to 5000 GPM.

3) The requirement that a precoat shall be prepared within an hour of use was removed. These statements now read "should" to allow a supervisor some discretion in the event a delay occurs.

4) All references to CP/0/A/8800/05 and CP/0/B/8800/04 were changed to reference Chemistry Management Procedure 3.4.17. These procedures were deleted and their contents incorporated in CMP 3.4.17.

5) Statements were added to explain that fresh resin beds can decrease the inventory of chemical additives in the secondary system. This is to alert operators to be aware of possible changes when placing CPDs in service.

6) A new enclosure was added to provide guidance for the inspection process when tube bundles are changed out.

7) The cleaning of CPD Y-Strainers was made into separate enclosures, and the process was expanded to be more detailed. This revision allows the job to be done with the vessel in "HOLD".

8) The process of placing a CPD in "HOLD" was made into separate enclosures. This operation can be routinely performed for a variety of reasons.

9) Steps were added throughout the procedure to require the review of Limits and Precautions and Log Books prior to performing any operations.

10) The requirement for a CPD to be backwashed prior to preparing filter media was added.

11) Guidance has been added for the use of premixed resins. Historically, we have mixed resins ourselves and this is still included. The development of the pre-mixed resins appear to have some attractive advantages. The General Office has requested that Catawba test some of these resins in the near future.

12) Guidance on the addition of Morpholine to CPD resins is included. Catawba will begin to add Morpholine to the secondary system in the near future. It is desirable to saturate the resin bed with Morpholine prior to placing it in service to reduce the chance of decreasing the system's chemical inventory. General Office guidance recommends a maximum addition of three gallons of 40% Morpholine to the Recirculation Tank.

13) A change was added to require recording the date a CPD is placed in service. This will aid in tracking the performance of the CPD and provide greater control over monitoring Secondary Chemistry.

14) The requirement to verify the availability of Demineralized Water (YM) was added in several sections.

15) All references to "Secondary Chemistry Supervisor" were changed to "Chemistry Supervisor".

16) All numerical ranges throughout the procedure were changed from a " ... +/- ..." format to "... to ..." format. This is consistent with procedure development guidelines.

17) References to specific sections of OP/0/A/6250/16 were made less specific. This will reduce the chances of changes made necessary to this procedure by a correspondence change in the referenced procedure.

18) Instructions were added to record specific information concerning CPD backwashes. This will aid in monitoring CPD performance and resin inventory in storage tanks.

19) The valve checklists were changed to delete YM valves and add several CM valves which were omitted in previous revisions.

20) Statements were added throughout the procedure to inform Operations when CPD valve manipulations could cause system flow, differential pressure, or bypass valve changes.

21) Electrical breaker lists were revised to match the in-plant naming of breakers.

22) The instructions for using/applying resin overlays were expanded.

23) The sections for preparing CPDs for maintenance has been totally rewritten. A vent/drain path has been incorporated into the process and is in place during maintenance activities.

24) The procedure sections have been renumbered extensively to reflect the additions/deletions.

25) References to specific sections of the procedure have been changed to reflect the renumbering that occurred.

PT/1/A/4200/136, Change #22

This change adds valve 1NI-136B to the Safety Injection (NI) cold shutdown procedure. This valve was previously tested quarterly under PT/1/A/4200/13A. It was discovered that opening 1NI-136B during power operation could degrade the Residual Heat Removal (ND) system flow in the event of a Large Break LOCA. Therefore, stroke testing of 1NI-136B will only be performed while Unit 1 is in modes 5, 6, or no mode. No unreviewed safety question is created by this change.

TN/1/A/0318/00/01A, Initial Issue

Implementation Procedure TN/1/A/0318/00/01A provides guidance for the installation of new piping and an ultrasonic flow meter for the Nuclear Service Water (RN) supply to Radiation Monitor 1EMF45A. The work associated with this TN is scheduled to occur during the Unit 1 outage while 'A' Train of the RN system supplying the associated Containment Spray (NS) heat exchanger is drained and isolated. The system alignment and Unit 1 Mode during this 'A' Train RN outage window make it possible to perform the work outlined in this implementation procedure without entering any kind of Technical Specification Action step. Both NS and the RN supply to the 'A' Train NS heat exchanger are not required to be operable during the time frame that this procedure is scheduled to be worked. There are no unusual RN system isolations planned to support implementation of this TN. Radiation Monitor 1EMF45A is not a component that is referenced in Tech. Specs. Therefore, isolation of this monitor will not cause any operability concerns. The electrical isolations associated with removing 1RNFS4980 and installing 1RNFE4980 and 1RNFT4980 only involve pulling fuses located in the EMF junction box. These electrical isolations will disable the local and Control Room alarms for 1EMF45A low flow indication. Again, since 1EMF45A is not a Tech. Spec. monitor, isolation of its low flow alarm indication is not an operability problem. The post-modification testing associated with this TN involves checking flow rates through 1EMF45A and verifying correct low flow annunciator/alarm response. These tests will occur with RN flow to 1EMF45A in an isolated, throttled, and full open alignment. The first two RN alignments are typical for normal system operation. The full open alignment merely involves opening the manual diaphragm valve located near the EMF. This is not considered to be an elaborate or highly unusual system alignment. Even in this atypical alignment, there would not be a significant reduction in RN flow to the NS heat exchanger. Operability of the NS heat exchanger would not be jeopardized in an unthrottled RN alignment to 1EMF45A. This implementation procedure, along with any other work control procedures, will adequately govern the return to service of all components/systems affected by this modification.

Based on the above discussion, it is concluded that there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/0318/00/02A, Initial Issue

Implementation Procedure TN/1/A/0318/00/02A provides guidance for the installation of new piping and an ultrasonic flow meter for the Nuclear Service Water (RN) supply to 1EMF45B. The work associated with this TN is

scheduled to occur during the Unit 1 outage while 'B' Train of the RN system supplying the associated Containment Spray (NS) heat exchanger is drained and isolated. The system alignment and Unit 1 Mode during this 'B' Train RN outage window make it possible to perform the work outlined in this implementation procedure without entering any kind of Technical Specification Action step. Both NS and the RN supply to the 'B' Train NS heat exchanger are not required to be operable during the time frame that this procedure is scheduled to be worked. There are no unusual RN system isolations planned to support implementation of this TN. Radiation Monitor 1EMF45B is not a component that is referenced in Tech. Specs. Therefore, isolation of this monitor will not cause any operability concerns. The electrical isolations associated with removing 1RNFS4990 and installing 1RNFE4990 and 1RNFT4990 only involve pulling fuses located in the EMF junction box. These electrical isolations will disable the local and Control Room alarms for 1EMF45B low flow indication. Again, since 1EMF45B is not a Tech. Spec. monitor, isolation of its low flow alarm indication is not an operability problem. The post-modification testing associated with this TN involves checking flow rates through 1EMF45B and verifying correct low flow annunciator/alarm response. These tests will occur with RN flow to 1EMF45B in an isolated, throttled, and full open alignment. The first two RN alignments are typical for normal system operation. The full open alignment merely involves opening the manual diaphragm valve located near the EMF. This is not considered to be an elaborate or highly unusual system alignment. Even in this atypical alignment, there would not be significant reduction in RN flow to the NS heat exchanger. Operability of the NS heat exchanger would not be jeopardized in an unthrottled RN alignment to 1EMF45B. This implementation procedure, along with any other work control procedures, will adequately govern the return to service of all components/systems affected by this modification.

Based on the above discussion, it is concluded that there are no unreviewed safety questions associated with the implementation of this procedure.

HP/0/B/1003/50, Initial Issue

Activities covered in this procedure concern the operation and calibration of personnel survey equipment, which will not increase the probability of an accident as described in Chapter 15 of the FSAR, nor create any situation that could cause an accident that has not already been evaluated in the FSAR. This procedure does not create an unreviewed safety question.

HP/0/B/1003/44, Initial Issue

Activities covered in this procedure concern the operation and calibration of personnel survey equipment, which will not increase the probability of an accident as described in Chapter 15 of the FSAR, nor create any situation that could cause an accident that has not already been evaluated in the FSAR. No unreviewed safety question is created by this procedure.

OP/2/A/6700/01, Change #129

OP/2/A/6700/01 (Unit Two Data Book) Table 2.2 is a table of data for use by plant personnel. The following describes the two sections of the table and how the data is obtained and used. Change #129 only affects the first section (power range full power currents). This change updates the full power currents determined by PT/2/A/4600/05D, Interim Incore/Excore Calibration. This procedure was performed to renormalize tilts to approximately 1.00, as Quadrant 4 lower tilt ratio was approaching 1.02.

FULL POWER CURRENTS

This section is used to record the 100% Full Power currents for +20, 0, and -20% axial offset and the M factors for each of the Power Range Excore Detectors. Data is obtained for this section only by use of approved procedures such as:

PT/2/A/4600/05A, Incore/Excore Calibration,
PT/2/A/4600/05D, Interim Incore/Excore Calibration, or
PT/2/A/4600/05G, Post Refueling Incore/Excore Calibration.

Other tests may supply data to this section, but in all cases the tests must be approved tests.

The data recorded here is used by Instrument and Electrical (IAE) personnel to adjust the Axial Flux Difference (AFD) calculating circuitry and inputs to Operator Aid Computer (OAC) programs. It is also used in manual calculations of AFD and Quadrant Power Tilt Ratio (QPTR) (for instance, when the OAC is inoperable).

Since AFD is used to dynamically adjust the Overtemperature Differential Temperature (OTDT) setpoints, the data here is safety related.

TRIP SETPOINTS

The trip setpoints for both the Intermediate (N-35 and N-36) and Power Range (N-41, N42, N43, and N44) detectors are recorded on page 2 of the table. These are used by IAE in setting the Reactor Trip setpoints for the detectors. The data here may be obtained by a variety of means.

Trip setpoints for Intermediate Range Detector may only be obtained by use of approved procedures to calculate or measure the 25% Full Power Reactor Trip setpoints.

Trip setpoints for the Power Range Detectors may not be deliberately set greater than 109% Full Power. The trip setpoint may be set lower than 109% by use of approved procedures, by direction of Tech. Specs., or by direction of the Shift Supervisor. The 109% is set by Tech. Spec. to ensure operation is bounded by the assumptions used in the FSAR chapter 15 accidents. Any setpoint below 109% may be used for conservatism or to comply with Tech. Specs.

Information in OP/2/A/6700/01 (Unit Two Data Book) is changed only by approved procedure change. It will not increase the probability/consequences of an accident analyzed in the FSAR or create an accident not analyzed in the FSAR. No analyzed or unanalyzed malfunction of safety related equipment will be created. The margin of safety as defined in the bases to Tech. Specs. will not be reduced in any way. There is no unreviewed safety question created by this change.

MP/0/A/7450/26, Change #3

The following changes have been made to the procedure:

- The bearing clearance reduction listed in Step 11.4.3.1 was changed to 0.002 to 0.003 to reflect vendor manual requirements.

- Step 11.4.3.6.1 was added to address the inside bearing replacement using Steps 11.4.3 through 11.4.3.6. Also, notes were added to this step about using Neolube on the tapered sleeve and about centering bearing in the bearing housing on the floating end of the fan shaft.

- Step 11.4.3 was further clarified by adding (Drive end).

- A sign off was added to the Data Sheets (Enclosure 13.1) to provide a sign off for the inside bearing cap torque in Step 11.4.3.4.

The changes being made to this procedure have been reviewed against approved vendor manuals, design documents, and station procedures to ensure that the corrective maintenance controlled by this procedure will return the fan to as-designed/as-built condition. These actions will ensure the fan's compliance with FSAR accident analysis. Since the fans will be returned to as-designed conditions, the possibility, consequences, or probability of a malfunction will be reduced. Therefore, no Unreviewed Safety Question exists.

Procedure MP/0/A/7650/63, Retype, Changes 0 to 2 Incorporated

This procedure provides a method for controlling vendor on-line leak sealing repairs. The procedure ensures that proper reviews and controls are taken for each leak sealing repair. The procedure also contains information specific to Catawba for tracking and trending.

The purpose of this evaluation is to describe the changes made to MP/0/A/7650/63 as part of a procedure rewrite. These changes are needed to provide additional documentation that vendor repairs are made following the specifications given in the procedure. No technical information was changed.

The following is a summary of the changes made to this procedure.

* The addition of documentation for the vendor repair.

  Added steps: 11.2 - Vendor documentation of adherence to procedural specifications.

11.3 - Vendor documentation of injection valve thread engagement

Revised steps: 11.4 - Vendor Documentation of injection process enhanced

The completion of this procedure requires that a safety evaluation (10CFR50.59) be completed and reviewed for each specific on-line leak sealing repair; therefore, this procedure does not significantly affect structures, systems, or components without a safety evaluation.

PT/2/A/4200/09A, Retype, Changes 0 to 70 Incorporated

This procedure has been completely rewritten so that the procedure format for both the Unit 1 and Unit 2 Auxiliary Safeguards Test Cabinet Periodic Test is the same. The testing methodology remains the same. The only changes made in the test are that the Auxiliary Feedwater (CA) Pumps will now be started in Sections 12.13 and 12.14, and a slave relay test device will not be placed across the contacts for the Main Feedwater (CF) Feed Regulating Bypass Valves unless the valves cannot be closed. In Sections 12.13 and 12.14, the CA pumps will be run in miniflow, since the discharge valves to the Steam Generators will be closed for the test. In the Feedwater Isolation sections, the CF Feed Regulating Bypass Valves will be verified to be closed prior to performing the test. If the valves cannot be closed, a slave relay test device will be used so that the circuit can be tested. Other than these changes, only the format of the test has been changed. No Unreviewed Safety Question is created by these changes.

MP/0/A/7450/47, Initial Issue

This safety evaluation is for the original issue of MP/0/A/7450/47.

This procedure has been written to provide guidance in performing corrective maintenance on the Chilled Water (YC) Chillers. Maintenance performed under this procedure has been reviewed against approved vendor manuals, design documents, and station procedures to ensure that corrective maintenance controlled by this procedure will return this chiller to as-built/as-designed condition. These actions will ensure the chiller's compliance with FSAR accident analysis. Since the chiller will be returned to as-designed conditions, the possibility, consequences or probability of a malfunction will be reduced. Therefore, no unreviewed safety question exists.

TN/1/A/1241/00/01A, Initial Issue

NSM CN-11241, Rev. 0, will revise the controls on the VF (Fuel Building Ventilation) system so that when a trip 2 signal is received from Radiation Monitors (EMFs) 35, 36, 37, and 42, the system will go into filter mode instead of shutting down the fans. This procedure will provide guidelines for isolation, electrical work, restorations, and functional verification of the modification.

This procedure may be implemented with Unit One in any mode. The isolations that will have to be performed to implement this procedure will effect the VF System (Fuel Building Ventilation), VE System (Annulus Ventilation), and two of the Reactor Vessel Head Vent valves.

During the implementation of this procedure, as a result of the isolations, both trains of VF will be inoperable. Per Tech. Spec. 3.9.11, this is allowed as long as no operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool is taking place. No fuel movement will be taking place during the implementation of this procedure.

Also, as a result of the isolations, VE Train A will be inoperable due to power being removed from VE Fan 1A, damper 1AVS-D-4 (1VE-4, VE Fan 1A Miniflow Isolation), and 1AVS-D-5 (VE Fan 1A Isolation Damper). Per Tech. Spec. 3.6.1.8, one train of VE may be inoperable for 7 days while the Unit is in Modes 1, 2, 3, or 4.

The isolations for this procedure affect Tornado Isolation Dampers 1FPX-D-3A, 1FPX-D-6A, 1FPX-D-3B, 1FPX-D-6B, and 1AVS-D-5. As a result of the isolations, Fuel Pool Exhaust dampers 1FPX-D-3A, 1FPX-D-6A, 1FPX-D-3B, and 1FPX-D-6B will fail open. However, these dampers have a back-up air supply tank that would close these dampers if the Tornado Isolation pushbutton was depressed on Control Board 1MC5. Damper 1AVS-D-5 is also a fail open damper, so this damper will be gagged closed before the isolations are made to protect the plant in case of a tornado. VE Train A will be inoperable during the implementation of this procedure, so having 1AVS-D-5 closed will not be an operational concern.

The isolations made for the Control Board plug P-04-16 also affect Reactor Head Vent valves 1NC251B and 1NC252B. Two of the four Reactor Head Vent valves, 1NC252B and 1NC253A (in series), are closed with power removed during normal operation. Therefore, a single failure would affect only one of the two powered valves, 1NC250A and 1NC251B. Valves 1NC251B and 1NC252B will have power removed from them to implement this procedure. 1NC251B will be tagged closed for this procedure, making one of the reactor coolant vent paths inoperable. Per Tech. Spec. 3.4.11, one reactor coolant vent path may be inoperable for 30 days while the Unit is in Modes 1, 2, 3, or 4.

All modified circuits will be functionally tested prior to returning them to service.

Accordingly, the implementation of this procedure will not increase the probability or consequences of an accident previously evaluated, or different than any already evaluated, in the FSAR. Nor will the implementation of this procedure increase the probability or consequences of an equipment malfunction previously evaluated, or different than any already evaluated, in the FSAR. The margin of safety defined in the bases of the Technical Specifications is unaffected, and no unreviewed safety questions exist.

OP/1/A/6350/02, Retype #15, Change 61 Incorporated

This evaluation for OP/1/A/6350/02, DIESEL GENERATOR OPERATION, documents the review of the change describing the inoperability of the Diesel Generators

(D/Gs) when performing an air roll or barring the engine. Additional changes are also incorporated and added to the retype.

The changes are as follows:

ENCLOSURES 4.2., 4.3, 4.4, 4.5

Complete retype of these valve checklists to add valve locations, reorganize the lists in order of room locations for the valves, and to incorporate new valves, and valve name changes.

ENCLOSURE 4.10

PROCEDURE

Steps 2.9 Deleted note concerning stopping the flow of oil to the turbochargers through the turbo oil solenoid valve after 5 minutes because it was already covered by a caution step.

Step 2.18 Moved note to reflect that the D/G is in operation and that data should be taken on PT/2/A/4350/10. It was previously after the D/G was paralleled, but data on the start was missed if the D/G was run unloaded for a time.

Step 2.20 Added range for acceptable D/G voltage before paralleling the D/G.

Step 2.22 Added the approximate rate of rotation for the synchroscope to the procedure to assist in the prevention of missed attempts to parallel the D/Gs.

Step 2.29 Added range for acceptable D/G voltage before paralleling the D/G.

Step 2.30 Added the approximate rate of rotation for the synchroscope to the procedure to assist in the prevention of missed attempts to parallel the D/Gs.

Step 2.45 Revised the step to include that the Effective Full Power Hour (EFPH) sheet has been included in the D/G Operating Parameters Periodic Test (PT) so that the EFPH data is stored with the D/G run data.

ENCLOSURE 4.11

PROCEDURE

Step 2.9 Deleted note concerning stopping the flow of oil to the turbochargers through the turbo oil solenoid valve after 5 minutes because it was already covered by a caution step.

Step 2.12 Moved note to reflect that the D/G is in operation and that data should be taken on PT/2/A/4350/10. It was previously after the

D/G was paralleled, but data on the start was missed if the D/G was run unloaded for a time.

Step 2.14 Added range for acceptable D/G voltage before paralleling the D/G.

Step 2.16 Added the approximate rate of rotation for the synchroscope to the procedure to assist in the prevention of missed attempts to parallel the D/Gs.

Step 2.23 Added range for acceptable D/G voltage before paralleling the D/G.

Step 2.24 Added the approximate rate of rotation for the synchroscope to the procedure to assist in the prevention of missed attempts to parallel the D/Gs.

Step 2.37 Revised the step to include that the Effective Full Power Hour sheet has been included in the D/G Operating Parameters PT so that the EFPH data is stored with the D/G run data.

ENCLOSURE 4.13

INITIAL CONDITIONS

Step 2.10 Changed step 2.6.1 to 2.7.1 to correct the wrong step number referenced.

Step 2.11 Changed step 2.6.3 to 2.7.3 to correct the wrong step number referenced.

ENCLOSURE 4.14

PROCEDURE

Rewrote the entire procedure section of the enclosure because the old enclosure left too many N/As for the startup portion of the procedure. The new enclosure consists of two options, each with the appropriate substeps to secure the D/G after an autostart.

ENCLOSURE 4.19

INITIAL CONDITIONS

Step 1.2 (old) Was deleted and a new Step 1.3 added to direct that the D/G must have been removed from service per Enclosure 4.13.

PROCEDURE

Step 2.1 Revised step to ensure that the D/G is in Maintenance mode.

Step 2.20 Revised step to return to Enclosure 4.13 to return the D/G to operability if it is the appropriate time to do so, and deleted

the remaining steps concerned with verifying the D/G is operable because they are covered in Enclosure 4.13.

ENCLOSURE 4.21

INITIAL CONDITIONS

Step 1.3 Note Changed the pressure where the D/G becomes inoperable to 210 psig based on the results of PIR 1-C90-0238 and the associated Tech. Spec. Interpretation revision 1 for Tech. Spec. 3.8.1.1.

PROCEDURE

Step 2.1 NOTE Revised note to allow the Operator to leave the room after returning all valves to the normal position, but before removing the crosstie hose.

ENCLOSURE 4.22

INITIAL CONDITIONS

Step 1.3 Note Changed the pressure where the D/G becomes inoperable to 210 psig based on the results of PIR 1-C90-0238 and the associated Tech. Spec. Interpretation revision 1 for Tech. Spec. 3.8.1.1.

PROCEDURE

Step 2.1 NOTE Revised note to allow the Operator to leave the room after returning all valves to the normal position, but before removing the crosstie hose.

These procedure changes have been evaluated to have no detrimental effect on plant safety, and create no unreviewed safety questions.

OP/2/A/6350/02, Retype #10, Changes 45 to 49 Incorporated

This evaluation for OP/2/A/6350/02, DIESEL GENERATOR OPERATION documents the review of the change in engine fluid temperatures deemed acceptable during the performance of the Engineered Safeguards (ES) checklist. Additional changes are also incorporated and added to the retype.

The changes are as follows:

LIMITS AND PRECAUTIONS

Step 2.3 Lube oil and cooling water at the engine outlets should be (140-150°F) while in standby unless the engine has been secured within the past 12 hours.*

ENCLOSURES 4.2, 4.3, 4.4, 4.5

Complete retype of these valve checklists to add valve locations, reorganize the lists in order of room locations for the valves, and to incorporate new valves, and valve name changes.

ENCLOSURE 4.6

Step 2 Jacket Water temp. maintained at (140-150°F) (as indicated by Strip Chart Recorder or Temp. Scanner on Diesel Generator (D/G) Control Panel 2A) unless the D/G has been secured within the last 12 hours. (The range becomes (140-185°F.)*

Step 3 Lube oil temp. maintained at (140-150°F) (as indicated by Strip Chart Recorder or Temp. Scanner on D/G Control Panel 2A) unless the D/G has been secured within the last 12 hours. (The range becomes (140-185°F.)*

ENCLOSURE 4.7

Step 2 Jacket Water temp. maintained at (140-150°F) (as indicated by Strip Chart Recorder or Temp. Scanner on Diesel Generator (D/G) Control Panel 2A) unless the D/G has been secured within the last 12 hours. (The range becomes (140-185°F.)*

Step 3 Lube oil temp. maintained at (140-150°F) (as indicated by Strip Chart Recorder or Temp. Scanner on D/G Control Panel 2A) unless the D/G has been secured within the last 12 hours. (The range becomes (140-185°F.)*

ENCLOSURE 4.8

Step 2 Jacket Water temp. maintained at (140-150°F) (as indicated by Strip Chart Recorder or Temp. Scanner on Diesel Generator (D/G) Control Panel 2B) unless the D/G has been secured within the last 12 hours. (The range becomes (140-185°F.)*

Step 3 Lube oil temp. maintained at (140-150°F) (as indicated by Strip Chart Recorder or Temp. Scanner on D/G Control Panel 2B) unless the D/G has been secured within the last 12 hours. (The range becomes (140-185°F.)*

ENCLOSURE 4.9

Step 2 Jacket Water temp. maintained at (140-150°F) (as indicated by Strip Chart Recorder or Temp. Scanner on Diesel Generator (D/G) Control Panel 2B) unless the D/G has been secured within the last 12 hours. (The range becomes (140-185°F.)*

Step 3 Lube oil temp. maintained at (140-150°F) (as indicated by Strip Chart Recorder or Temp. Scanner on D/G Control Panel 2B) unless

the D/G has been secured within the last 12 hours. (The range becomes (140-185°F.)*

ENCLOSURE 4.10

INITIAL CONDITIONS

Step 1.3 Lube Oil temperature and jacket water temperature are (140-150°F) unless the D/G has secured within the last 12 hours. (The range becomes (140-185°F.)*

PROCEDURE

Step 2.9 Deleted note concerning stopping the flow of oil to the turbochargers through the turbo oil solenoid valve after 5 minutes because it was already covered by a caution step.

Step 2.18 Moved note to reflect that the D/G is in operation and that data should be taken on PT/2/A/4350/10. It was previously after the D/G was paralleled, but data on the start was missed if the D/G was run unloaded for a time.

Step 2.20 Added range for acceptable D/G voltage before paralleling the D/G.

Step 2.22 Added the approximate rate of rotation for the synchroscope to the procedure to assist in the prevention of missed attempts to parallel the D/Gs.

Step 2.29 Added range for acceptable D/G voltage before paralleling the D/G.

Step 2.30 Added the approximate rate of rotation for the synchroscope to the procedure to assist in the prevention of missed attempts to parallel the D/Gs.

Step 2.45 Revised the step to include that the Effective Full Power Hour (EFPH) sheet has been included in the D/G Operating Parameters Periodic Test (PT) so that the EFPH data is stored with the D/G run data.

ENCLOSURE 4.11

INITIAL CONDITIONS

Step 1.3 Lube Oil temperature and jacket water temperature are (140-150°F) unless the D/G has been secured within the last 12 hours. (The range becomes (140-185°F.)*

PROCEDURE

Step 2.9 Deleted note concerning stopping the flow of oil to the turbochargers through the turbo oil solenoid valve after 5 minutes because it was already covered by a caution step.

Step 2.12 Moved note to reflect that the D/G is in operation and that data should be taken on PT/2/A/4350/10. It was previously after the D/G was paralleled, but data on the start was missed if the D/G was run unloaded for a time.

Step 2.14 Added range for acceptable D/G voltage before paralleling the D/G.

Step 2.16 Added the approximate rate of rotation for the synchroscope to the procedure to assist in the prevention of missed attempts to parallel the D/Gs.

Step 2.23 Added range for acceptable D/G voltage before paralleling the D/G.

Step 2.24 Added the approximate rate of rotation for the synchroscope to the procedure to assist in the prevention of missed attempts to parallel the D/Gs.

Step 2.37 Revised the step to include that the Effective Full Power Hour sheet has been included in the D/G Operating Parameters PT so that the EFPH data is stored with the D/G run data.

*This range was increased from (140-150°F) to (140-185°F) because the upper range did not accurately reflect true conditions of the D/G when performing an ES checklist immediately after shutting down the Diesel. The range also did not accurately reflect the requirements established for performing a D/G restart shortly after securing the D/G.

The previous ranges were set in place to verify that the D/G Jacket Water Keep Warm system and D/G Lube Oil Prelube system were functioning properly. The lower ranges assured that the jacket water and lube oil heaters were energizing at the proper temperature to maintain the D/Gs in a ready to start condition. The upper ranges assured that the temperatures of the lube oil were not excessive for long periods of time, breaking down the oil at a higher than necessary rate. Both of these parameters are still being met by the D/G procedures. During times of standby alignment, when the D/G has not been run within 12 hours, the acceptable range of jacket water and lube oil temperatures remains at the (140-150°F) range. Only after running the D/Gs and then reading the temperatures do the new (140-185°F) ranges apply.

Making this change of acceptable temperatures decreases the time that each D/G is declared inoperable. Currently, Operations personnel are waiting several hours after each D/G run to declare the D/G operable because the engine has not cooled from the normal operating temperatures of 185°F to the normal standby temperature ranges. This delay does not enhance plant safety because the D/G has just proven its ability to start, accelerate, and carry its assigned load. D/G temperatures decrease from normal operating to standby at their given rate without regard to D/G operability.

Should elevated temperatures due to improper heater operation be present, they would be noted out of range during the next Operator semi-daily rounds of the D/G room. Corrective actions would then be taken per normal practice.

ENCLOSURE 4.13

INITIAL CONDITIONS

Old Step 1.3 Was deleted concerning the Unit Supervisor's permission to remove the D/G from service because the job is ordered by the Unit Supervisor.

Steps 1.4, 1.5 Added the steps to give pre-conditions to D/G removal.

Step 1.6 Added Tech. Spec. 3.8.1.2 to the initial conditions for Mode 5 and 6 operation. Tech. Specs. 3.1.2.1 and 3.1.2.2 require an operable power source, but 3.8.1.2 requires an operable D/G to back up the power source during core alterations of Mode 6. The additional Tech. Spec. has the Operators look at all the Mode 6 requirements.

PROCEDURE

Step 2.1 Changed the note reminding the Operator to do PT/2/A/4350/02C to a sign-off step.

Step 2.10 Changed step 2.6.1. to 2.7.1 to correct the wrong step number referenced.

Step 2.11 Changed step 2.6.3 to 2.7.3 to correct the wrong step number referenced.

ENCLOSURE 4.14

PROCEDURE

Rewrote the entire procedure section of the enclosure because the old enclosure left too many N/As for the startup portion of the procedure. The new enclosure consists of two options, each with the appropriate substeps to secure the D/G after an autostart.

ENCLOSURE 4.18

INITIAL CONDITIONS

Step 1.3 Added that the opposite train D/G supplied components must be operable to prevent simultaneous inoperability on both trains of equipment and entering Tech. Spec. 3.0.3.

PROCEDURE

Step 2.1 Re-wrote the step to acknowledge that the D/G is inoperable, but because the inoperability was short (approximately ten minutes), there is acceptably small risk associated with the D/G's inoperability. Therefore, as long as the inoperability does not approach the two hour Tech. Spec. Action, the air roll may be performed without performing plant realignments.

ENCLOSURE 4.19

INITIAL CONDITIONS

Step 1.2 (old) Was deleted and a new Step 1.3 added to direct that the D/G must have been removed from service per Enclosure 4.13.

PROCEDURE

Step 2.1 Revised step to ensure that the D/G is in Maintenance mode.

Step 2.20 Revised step to return to Enclosure 4.13 to return the D/G to operability if it is the appropriate time to do so, and deleted the remaining steps concerned with verifying the D/G is operable because they are covered in Enclosure 4.13.

ENCLOSURE 4.21

INITIAL CONDITIONS

Step 1.3 Note Changed the pressure where the D/G becomes inoperable to 210 psig based on the results of PIR 1-C90-0238 and the associated Tech. Spec. Interpretation revision 1 for Tech. Spec. 3.8.1.1.

PROCEDURE

Step 2.1 Note Revised note to allow the Operator to leave the room after returning all valves to the normal position, but before removing the crosstie hose.

ENCLOSURE 4.22

INITIAL CONDITIONS

Step 1.3 Note Changed the pressure where the D/G becomes inoperable to 210 psig based on the results of PIR 1-C90-0238 and the associated Tech. Spec. Interpretation revision 1 for Tech. Spec. 3.8.1.1.

PROCEDURE

Step 2.1 Note Revised note to allow the Operator to leave the room after returning all valves to the normal position, but before removing the crosstie hose.

ENCLOSURE 4.24

PROCEDURE

Step 2.1.7 Added the computer point number to monitor to observe Nuclear Service Water (RN) flow through the Diesel Generator Engine Cooling Water Heat Exchanger (KD Hx).

Step 2.3.7 Added the computer point number to monitor to observe RN flow through the KD Hx.

These procedure changes have been evaluated to have no detrimental effect on plant safety, and create no unreviewed safety questions.

PT/1/A/4450/03C, Retype, Changes 0 to 12 Incorporated

Section 12.1 and 12.5

Section 12.1 (Train A) and section 12.5 (Train B) of this procedure verify that the Annulus Ventilation System (VE) will draw-down the annulus to a pressure equal to or more negative than -0.5 inches of water gauge (inwg) within one minute after a start signal as required by Tech. Spec. 4.6.1.8.d.4. Also, the stabilized pressure and drawdown time are checked to verify that they are within the limits set by Design Engineering. If the stabilized pressure or the drawdown time are not within the above limits, Design Engineering is contacted within twelve hours to determine operability (if unit 1 is in mode 1 to 4). The test manometer vent is the only part of this section that could have an effect on the VE system. An isolation valve exists on the vent line between the pressure transmitter and the manometer, so that the manometer can be isolated easily from the transmitter if needed. A caution statement is included in the procedure that directs the performer to reconnect the manometer or close the isolation valve immediately if the manometer inadvertently becomes disconnected from the vent line. If an accident occurs during the performance of this section, VE would perform as designed.

Sections 12.2 and 12.6

Section 12.2 (Train A) and section 12.6 (Train B) of this procedure verify that the pressure drop across the total filter unit is less than 8 inwd and that the system flow rate is 9000 cfm ± 10% during system operation, as required by Tech. Specs. 4.6.1.8.b.3 and d.1. The upper limit of the flow is currently restricted to 8700 cfm by PIR #C90-00093, and this is reflected in the present acceptance criteria. All of the data in sections 12.2 and 12.6 is taken while the system is operating per OP/1/A/6450/02. None of the data taken will affect the system operation. Therefore, if an accident were to occur during the performance of these sections, VE would function as designed.

Sections 12.3 and 12.7

Section 12.3 (Train A) and section 12.7 (Train B) of this procedure verify that the pre-heaters dissipate 45 ± 6.7 kW during system operation as required by Tech. Spec. 4.6.1.8.d.5. The lower limit of the power dissipated is currently restricted to 45 kW by PIR #C90-00093 and the acceptance criteria reflects this restriction. The heater control panel is opened and then Amps and Volts on all three phases of both stages of the heater are measured using an ammeter and a multimeter. Calculations are performed to give total voltage adjusted (to 600V) power dissipation. These sections are performed while the system is operating per OP/1/A/6450/02 and have no effect on the system other than temporarily removing the power from the heater. If an accident occurs during performance of these sections, VE would function as designed.

Section 12.4 and 12.8

Section 12.4 (Train A) and section 12.8 (Train B) of this procedure verify that the filter cooling electric motor-operated bypass valves can be manually opened such that a mini-flow path is established from the non-operating filter train to the operating filter train as required by Tech. Spec. 4.6.1.8.d.3. This is accomplished by ensuring that the damper control switches for each trains' damper is in the auto position, verifying that the opposite trains' damper is open, then placing the switch for the applicable train damper to the "Open" position and verifying that the valve opens. The opposite train flow is recorded for information while the system is in the mini-flow alignment to verify cooling flow. Then the opposite train valve is closed by placing the switch to the "Close" position to verify proper operation of the damper control switch and to verify that the valve will close. All damper control switches are independently verified to be returned to the "As Found" position by the procedure. A caution statement gives the performer direction to isolate the mini-flow path in the event of an accident during the test because the operating filter unit flow is affected when in mini-flow alignment. These sections are performed while the system is operating per OP/1/A/6450/02. If an accident occurs during the performance of these sections, VE would function as designed.

Section 12.9, 12.10 and 12.11

Sections 12.9, 12.10 and 12.11 verify that the proper vacuum decay time can be attained for different alignments of VE components and other ventilation systems. These tests are performed as a result of testing conducted on the VE system during the Unit 2 End-of-Cycle 3 (2EOC3) Refueling Outage, and LER 414/90-03. The vacuum decay time test measures the time that it takes the annulus pressure to decay from -3.5 inches of water to - 0.5 inches of water with VE shutdown. The purpose of this test is to ensure less than 2,000 scfm air in-leakage into the Annulus under accident conditions. The 2000 scfm in-leakage is the assumed value used in CANVENT and ACTDOS to calculate dose for the post LOCA conditions. This test is not required directly by Tech. Spec., but in-leakage outside the above mentioned value may increase the dose above the appropriate 10CFR limits.

All of these sections shutdown both Trains of the Fuel Pool Ventilation System (VF) on Unit 1 per OP/1/A/6450/04. VF is only required to be operable during fuel movement or with any overhead load above the fuel pool. A prerequisite to the tests ensure that there is no work in progress or planned during the performance of these tests that may require the operability of VF on Unit 1. After the testing is complete, VF is returned to normal per OP/1/A/6450/04. Unit 1 is also required to be in modes 5, 6 or no mode during the performance of these sections. VE is not required to be operable during modes 5, 6 or no mode, so these test could not increase the consequences of an accident or decrease the margin of safety as defined in the bases of Tech. Specs. Various backdraft dampers are tied open in sections 12.9 and 12.11. Because of the possibility of affecting the flow when a damper is tied open, the flow for each train is required to be checked after the dampers have been returned to normal and before Unit 1 enters mode 4 (sections 12.9 and 12.11 only). The unit vent isolation dampers (D-5 & D-10) are aligned using the damper control switches in the control room. These switches are returned to the "Auto" Position by the procedure when testing is complete.

Section 12.9 performs the vacuum decay time test with dampers D-5, D-10, D-11 and D-12 open to verify annulus pressure boundary integrity during mini-flow operation of either VE train. Section 12.10 performs the vacuum decay test with dampers D-5 and D-10 open and a door on the B train filter unit open (randomly chosen). This test verifies the annulus pressure boundary during normal maintenance activities that may be performed when Unit 1 is in Mode 1 to 4.

Section 12.11 will perform the Vacuum Decay Time Test under accident conditions for the Auxiliary Building Ventilation System (VA) system with VE dampers D-1, D-6, D-11 and D-12 open. This test ensures integrity of the VE ductwork, filter units and the general Annulus while VA is in the most conservative accident alignment. During this test, Train 1B & 2B of VA will be shutdown per OP/0/A/6450/03 and train 1A & 2A filtered exhaust fans will be placed in the post LOCA mode of operation by this procedure. A prerequisite requires that Radiation Protection is informed of the intended abnormal alignment of VA and that exhaust flow from the sample hoods will be lost. After the test is completed, VA is returned to normal per Operations procedure OP/0/A/6450/03. If an accident were to occur on the opposite Unit during the performance of section 12.11, VA would function as designed because it would already be in the safety alignment.

Summary

All instrumentation installed/removed, jumpers placed/removed, and transmitters failed and returned to service are independently verified within each section of this procedure. The VE, VF, and VA systems are operated under the appropriate Operations procedures. Where applicable. caution statements are included to ensure that the system is returned to normal in a timely manner in the event of an accident during the performance of this procedure, so the margin of safety as defined in the bases of Tech. Spec. will not be reduced. VE is not an accident initiator, so the probability of an accident will not be increased, and the possibility of an accident different than evaluated will not be created. For these reasons, and the ones stated within this evaluation, this procedure does not increase the consequences of an accident OR increase the probability or consequences of a malfunction of equipment important to safety. Further, the possibility of a malfunction of equipment important to safety different than evaluated will not be created. Therefore, an unreviewed safety question does not exist.

PT/2/A/4450/03C, Retype, Changes 0 to 9 Incorporated

Section 12.1 and 12.5

Section 12.1 (Train A) and section 12.5 (Train B) of this procedure verify that the Annulus Ventilation System (VE) will draw-down the annulus to a pressure equal to or more negative than -0.5 inches of water gauge (inwg) within one minute after a start signal as required by Tech. Spec. 4.6.1.8.d.4. Also, the stabilized pressure and drawdown time are checked to verify that they are within the limits set by Design Engineering. If the stabilized pressure or the drawdown time are not within the above limits, Design Engineering is contacted within twelve hours to determine operability (if unit 2 is in mode 1 to 4). The test manometer vent is the only part of this

section that could have an effect on the VE system. An isolation valve exists on the vent line between the pressure transmitter and the manometer, so that the manometer can be isolated easily from the transmitter if needed. A caution statement is included in the procedure that directs the performer to reconnect the manometer or close the isolation valve immediately if the manometer inadvertently becomes disconnected from the vent line. If an accident occurs during the performance of this section, VE would perform as designed.

Sections 12.2 and 12.6

Section 12.2 (Train A) and section 12.6 (Train B) of this procedure verify that the pressure drop across the total filter unit is less than 8 inwd and that the system flow rate is 9000 cfm ± 10% during system operation, as required by Tech. Specs. 4.6.1.8.b.3 and d.1. The upper limit of the flow is currently restricted to 8700 cfm by PIR #C90-00093, and this is reflected in the present acceptance criteria. All of the data in sections 12.2 and 12.6 is taken while the system is operating per OP/2/A/6450/02. None of the data taken will affect the system operation. Therefore, if an accident were to occur during the performance of these sections, VE would function as designed.

Sections 12.3 and 12.7

Section 12.3 (Train A) and section 12.7 (Train B) of this procedure verify that the pre-heaters dissipate 45 ± 6.7 kW during system operation as required by Tech. Spec. 4.6.1.8.d.5. The lower limit of the power dissipated is currently restricted to 45 kW by PIR #C90-00093, and the acceptance criteria reflects this restriction. The heater control panel is opened and then Amps and Volts on all three phases of both stages of the heater are measured using an ammeter and a multimeter. Calculations are performed to give total voltage adjusted (to 600V) power dissipation. These sections are performed while the system is operating per OP/2/A/6450/02 and have no effect on the system other than temporarily removing the power from the heater. If an accident occurs during performance of these sections, VE would function as designed.

Section 12.4 and 12.8

Section 12.4 (Train A) and section 12.8 (Train B) of this procedure verify that the filter cooling electric motor-operated bypass valves can be manually opened such that a mini-flow path is established from the non-operating filter train to the operating filter train as required by Tech. Spec. 4.6.1.8.d.3. This is accomplished by ensuring that the damper control switches for each trains' damper is in the auto position, verifying that the opposite trains' damper is open, then placing the switch for the applicable train damper to the "Open" position and verifying that the valve opens. The opposite train flow is recorded for information while the system is in the mini-flow alignment to verify cooling flow. Then the opposite train valve is closed by placing the switch to the "Close" position to verify proper operation of the damper control switch and to verify that the valve will close. All damper control switches are independently verified to be returned to the "As Found" position by the procedure. A caution statement gives the performer direction to isolate the mini-flow path in the event of an accident during the test because the operating filter unit flow is affected when in mini-flow alignment. These sections are performed while the system is operating per OP/2/A/6450/02. If

an accident occurs during the performance of these sections, VE would function as designed.

Section 12.9, 12.10 and 12.11

Sections 12.9, 12.10 and 12.11 verify that the proper vacuum decay time can be attained for different alignments of VE components and other ventilation systems. These tests are performed as a result of testing conducted on the VE system during the Unit 2 End-of-Cycle 3 (2EOC3) Refueling Outage, and LER 414/90-03. The vacuum decay time test measures the time that it takes the annulus pressure to decay from -3.5 inches of water to - 0.5 inches of water with VE shutdown. The purpose of this test is to ensure less than 2,000 scfm air in-leakage into the Annulus under accident conditions. The 2000 scfm in-leakage is the assumed value used in CANVENT and ACTDOS to calculate dose for the post LOCA conditions. This test is not required directly by Tech. Spec., but in-leakage outside the above mentioned value may increase the dose above the appropriate 10CFR limits.

All of these sections shutdown both Trains of the Fuel Pool Ventilation System (VF) on Unit 2 per OP/2/A/6450/04. VF is only required to be operable during fuel movement or with any overhead load above the fuel pool. A prerequisite to the tests ensure that there is no work in progress or planned during the performance of these tests that may require the operability of VF on Unit 2. After the testing is complete, VF is returned to normal per OP/2/A/6450/04. Unit 2 is also required to be in modes 5, 6 or no mode during the performance of these sections. VE is not required to be operable during modes 5, 6 or no mode, so these test could not increase the consequences of an accident or decrease the margin of safety as defined in the bases of Tech. Specs. Various backdraft dampers are tied open in sections 12.9 and 12.11. Because of the possibility of affecting the flow when a damper is tied open, the flow for each train is required to be checked after the dampers have been returned to normal and before Unit 1 enters mode 4 (sections 12.9 and 12.11 only). The unit vent isolation dampers (D-5 & D-10) are aligned using the damper control switches in the control room. These switches are returned to the "Auto" Position by the procedure when testing is complete.

Section 12.9 performs the vacuum decay time test with dampers D-5, D-10, D-11 and D-12 open to verify annulus pressure boundary integrity during mini-flow operation of either VE train. Section 12.10 performs the vacuum decay test with dampers D-5 and D-10 open and a door on the B train filter unit open (randomly chosen). This test verifies the annulus pressure boundary during normal maintenance activities that may be performed when Unit 2 is in Mode 1 to 4.

Section 12.11 will perform the Vacuum Decay Time Test under accident conditions for the Auxiliary Building Ventilation System (VA) system with VE dampers D-1, D-6, D-11 and D-12 open. This test ensures integrity of the VE ductwork, filter units and the general Annulus while VA is in the most conservative accident alignment. During this test, Train 1B & 2B of VA will be shutdown per OP/0/A/6450/03 and train 1A & 2A filtered exhaust fans will be placed in the post-LOCA mode of operation by this procedure. A prerequisite requires that Radiation Protection is informed of the intended abnormal alignment of VA and that exhaust flow from the sample hoods will be lost. After the test is completed, VA is returned to normal per Operations procedure

OP/0/A/6450/03. If an accident were to occur on the opposite Unit during the performance of section 12.11, VA would function as designed because it would already be in the safety alignment.

Summary

All instrumentation installed/removed, jumpers placed/removed, and transmitters failed and returned to service are independently verified within each section of this procedure. The VE, VF, and VA systems are operated under the appropriate Operations procedures. Where applicable, caution statements are included to ensure that the system is returned to normal in a timely manner in the event of an accident during the performance of this procedure, so the margin of safety as defined in the bases of Tech. Spec. will not be reduced. VE is not an accident initiator, so the probability of an accident will not be increased, and the possibility of an accident different than evaluated will not be created. For these reasons, and the ones stated within this evaluation, this procedure does not increase the consequences of an accident OR increase the probability or consequences of a malfunction of equipment important to safety. Further, the possibility of a malfunction of equipment important to safety different than evaluated will not be created. Therefore, an unreviewed safety question does not exist.

PT/1/A/4550/10, Retype #4, Changes 9 to 9 Incorporated

This revision of PT/1/A/4350/10, "Diesel Generator Operating Parameters," was revised to more closely reflect the actual order that the operating data was required to be obtained, and to match up more closely with OP/1/A/6350/02, "Diesel Generator Operating Procedure," for taking the data. Additionally, the revision contains an enclosure to calculate the Effective Full Power Hours of the Diesel Generator (D/G) for the run. This follows the precedent set up by the monthly D/G Periodic Test (PT), PT/1/A/4350/02A(B), which records this data with the run data and stores it with the permanent records. Additional changes were placed in the data collection sheets for a column to define which parameters could be adequately recorded with check marks and which required actual values to be recorded. This is to enhance the quality of the data trended by Maintenance Engineering Services (MES) for predicting engine problems.

The only other significant changes to the PT were the deletion of the data sheet enclosure due to a revision in the requirements for the sheet, and the addition of an enclosure to handle out of the normal D/G parameters before they reach the point of causing inoperability.

These changes have been evaluated by referencing Tech. Specs. 3/4.8.1.1, FSAR Section 8.3.1, 9.5.4, 9.5.5, 9.5.6 and 9.5.7. Based on the reviews of these documents, there were no safety related concerns created by these procedure changes. Additionally, no unreviewed safety questions were created by these changes.

PT/2/A/4550/10, Retype #4, Changes 9 to 9 Incorporated

This revision of PT/2/A/4350/10, "Diesel Generator Operating Parameters," was revised to more closely reflect the actual order that the operating data was required to be obtained, and to match up more closely with OP/2/A/6350/02, "Diesel Generator Operating Procedure" for taking the data. Additionally, the revision contains an enclosure to calculate the Effective Full Power Hours of the D/G for the run. This follows the precedent set up by the monthly D/G PT, PT/2/A/4350/02A(B), which records this data with the run data and stores it with the permanent records. Additional changes were placed in the data collection sheets for a column to define which parameters could be adequately recorded with check marks and which required actual values to be recorded. This is to enhance the quality of the data trended by MES for predicting engine problems.

The only other significant changes to the PT were the deletion of the data sheet enclosure due to a revision in the requirements for the sheet, and the addition of an enclosure to handle out of the normal D/G parameters before they reach the point of causing inoperability.

These changes have been evaluated by referencing Tech. Specs. 3/4.8.1.1, FSAR Sections 8.3.1, 9.5.4, 9.5.5, 9.5.6 and 9.5.7. Based on the reviews of these documents, there were no safety related concerns created by these procedure changes. Additionally, no unreviewed safety questions were created by these changes.

MP/0/A/7150/90, Initial Issue

This safety evaluation is for the creation of MP/0/A/7150/90. This procedure has been written to provide a method to test and document the condition of ice condenser ice basket U-bolts. The failure of ice basket U-bolts at McGuire has been well documented on their PIR number 0-M90-0289.

The test torque of 19 ft-lbs. was determined through an evaluation of design loads plus a factor in excess of 25% which includes test uncertainties, fatigue concerns, and a factor of safety. McGuire procedure MP/0/A/7150/99, "Ice Basket U-bolt Torque", was used to determine the nut torque that would produce the required test load on the bolts. It was determined that 19 ft-lbs. was the appropriate torque. Design has determined that this same torque can be used at Catawba. Refer to the Catawba Operability evaluation for PIR 0-C90-0322.

The second part of the procedure involves returning the U-bolt nuts to the as-built torque. The "Westinghouse Ice Basket Installation Procedure", our number CNM 1201.17-343, states that the proper nut torque is 12 ft-lbs. in Step 4.21.

This procedure has been written based on approved vendor manuals, design documents, and station procedures to ensure that the actions controlled will return the ice basket U-bolts to their as designed/as built condition following acceptance testing. These actions will ensure the ice condenser's compliance with FSAR accident analysis. Since the U-bolts will be returned as-designed conditions, the possibility, consequences, or probability of a

malfunction will be reduced. Therefore, no unreviewed safety questions exists.

PT/0/A/4150/12B, Initial Issue

Tech. Spec. 3.1.1.3 requires that within 7 Effective Full Power Days (EFPD) after reaching a boron concentration of 300 ppmB, the Moderator Temperature Coefficient (MTC) of Reactivity be verified to be less negative than 41 pcm/deg F. This procedure measures the MTC.

The MTC is calculated by measuring the Hot Full Power (HFP) Boron concentration at two different reactor coolant temperatures while maintaining the same power level and control rod position. The boron difference is converted into a reactivity difference, and the Isothermal Temperature Coefficient (ITC) calculated. From the ITC, the Doppler Temperature Coefficient is subtracted to give the MTC.

The procedure requires that 3 sets of steady state data (boron, temperature, and other reactivity parameters) be obtained. The first data set is obtained at normal full power conditions. The second is obtained after the Reactor Coolant (NC) system Average Temperature (T-AVG) has been lowered 6 Degrees F below Reference Temperature (T-REF) while maintaining power and control rod position the same as in the first data set. The NC system temperature is lowered by a controlled boration of the NC system. The third data set is obtained after increasing the NC system T-AVG to within 1 Degree F of T-REF by a controlled dilution of the NC system (data set is obtained with power and control rods the same as the first two data sets).

The only off normal condition during this test is the NC T-AVG being 6 Degrees F lower than T-REF. T-AVG is closely monitored throughout the test to ensure that the deviation between T-AVG/T-REF does not approach 7.8 Degrees F (the point at which the turbine control valves are at the full open position). The normal FSAR accident analysis assumes a 4 Degree F T-AVG/T-REF deviation (15.0.3.2), but deviations for required testing are allowed by FSAR 15.0.1.1. Therefore, the temperature deviations are bounded by the FSAR analysis.

An Axial Flux Difference oscillation is induced by the required temperature swings but is monitored closely and must remain within the operating windows.

This test does not impact any equipment important to safety or lower the margin of safety.

PT/1/A/4150/12B, Change #5

This procedure is being deleted. It is being replaced by a Unit "0" shared procedure, PT/0/A/4130/12B, "Moderator Temperature Coefficient of Reactivity Measurement (EOC)". PT/0/A/4150/12B is the same as this procedure except for deletion of unit specific identification and typographical corrections and clarifications. (See earlier summary for the new procedure.)

PT/2/A/4150/12B, Change #4

This procedure is being deleted. It is being replaced by a Unit "0" shared procedure, PT/0/A/4130/12B, "Moderator Temperature Coefficient of Reactivity Measurement (EOC)". PT/0/A/4150/12B is the same as this procedure except for deletion of unit specific identification and typographical corrections and clarifications. (See earlier summary for the new procedure.)

OP/1/A/6700/01, Change #195

This change replaces page 1 of Table 2.2. New Full Power currents and M_j factors were obtained from PT/1/A/4600/05A, "Incore/Excore Calibration." Also, the "C" Incore Detector Setpoints in Table 2.4 were replaced. During PT/1/A/4600/06B, "Incore Detector Setpoints Determination," it was discovered that Detector "C" cable was only 1771 inches long. This change blocks administratively path #4 (Normal Mode) which is greater than 1771 inches.

TABLE 2.4

The Movable Incore Detector System (ENA) is used to measure the flux distribution in the Reactor Core. While the system is not considered by the FSAR to be important to safety, the data gathered by the system is used for 3 Tech. Spec. related functions. These are:

1) Recalibration of the Excore Neutron Flux Detection System (inputs to Overtemperature Differential Temperature (OTDT), Overpower Differential Temperature (OPDT) and Axial Flux Difference (AFD))

2) Monitoring the Quadrant Power Tilt Ratio (if one Nuclear Instrumentation System channel is unavailable)

3) Measurement of $F^N_{\Delta H}$ and $F_Q(Z)$

Data Book Table 2.4 serves as a reference for the top and bottom of core setpoints used in the ENA system. The setpoints are calculated and measured by PT/1/A/4600/06B, "Incore Detector Setpoints Determination". The setpoints are used so that the ENA system correctly scans the entire height of the core. Without the correct setpoints, the ENA system would not correctly give a measurement of the flux distribution of the core.

This table of the data book does not impact any safety related equipment. It does not increase or create the possibility of any accident analyzed (or unanalyzed) by the FSAR. No safety margins are reduced.

TABLE 2.2

OP/1/A/6700/01 (Unit One Data Book) Table 2.2 is a table of data for use by plant personnel. The following describes the 3 sections of the table and how the data is obtained and used.

FULL POWER CURRENTS

This section is used to record the 100% Full Power Zero Axial offsets currents and M factors for each of the Power Range Excore (ENB) Detectors. Data is obtained for this section only by use of approved procedures such as PT/1/A/4600/05E, "Refueling ENB Calibration", or PT/1/A/4600/05A, "Incore/Excore Calibration." Other tests may supply data to this section, but in all cases the tests must be approved tests.

The data recorded here is used by Instrument and Electrical (IAE) to adjust the Axial Flux Difference (AFD) calculating circuitry and Operator Aid Computer (OAC) programs. It may also be used to manually calculate AFD if the OAC is inoperable.

Since AFD is used to dynamically adjust both the OTDT and OPDT setpoints, the data here is safety related. The accidents referenced above depend on these setpoints for mitigation.

TRIP SETPOINTS

The trip setpoints for both the Intermediate (N-35 and N-36) and Power Range (N-41, 42, 43 and 44) Detectors are recorded on page 2 of the table. These are used by IAE in setting the Reactor Trip setpoints on the detectors. The data here may be obtained by a variety of means.

Trip setpoints for Intermediate Range Detector may only be obtained by use of approved procedures to calculate or measure the 25% Full Power Reactor Trip setpoints.

Trip setpoints for the Power Range Detectors may not be deliberately set greater than 109% Full Power ever. The trip setpoint may be set lower than 109% by use of approved procedures, by direction of Tech. Specs. or by direction of the Shift Supervisor. The 109% is set by Tech. Spec. to ensure operation is bounded by the assumptions used in the FSAR chapter 15 accidents. Any setpoint below 109% may be used for conservatism or to comply with Tech. Specs.

Information in OP/1/A/6700/01 (Unit One Data Book) is changed only by approved procedure change. It will not increase the probability/consequences of an accident analyzed in the FSAR or create an accident not analyzed in the FSAR. No equipment other than the Nuclear Instrumentation System is affected by Table 2.2. The safety margin will not be decreased.

No unreviewed safety question is created by either of these changes.

PT/1/A/4200/09C, Change #2

The purpose of this test is to verify that each automatic valve actuated by the Boron Dilution Mitigation System (BDMS) moves to its correct position upon receipt of a trip signal and also verify that each reactor makeup water pump stops upon receipt of a trip signal.

Suction of the Centrifugal Charging (NV) pumps will be aligned to the Volume Control Tank (VCT) and the Reactor Makeup Water Pumps will be running in recirculation to the Reactor Makeup Water Storage Tank (RMWST). A Boron Dilution Mitigation System trip signal will be simulated by depressing the TEST pushbutton on the Shutdown Margin Monitor. Suction of the NV Pumps will be verified to swap to the Refueling Water Storage Tank (FWST) and the Reactor Makeup Water Pumps will trip.

The additional changes will allow testing in modes 3 and 4 when the NV pump is in operation, whereas the original procedure was to be performed while the NV pumps were off. The NV pump is providing seal injection to the Reactor Coolant pump seals. These changes will cause a swap of suction sources from the VCT to the FWST while the NV pump is operating. This change of suction sources with the pump operating is the original design of the system; therefore, no new equipment malfunction is created; thus, the consequences of equipment malfunction remains the same.

During this test, one train of the BDMS will remain in service. The opposite train will be inoperable for a period of time well within the action time allowed by Tech. Specs.; therefore, the margin of safety as defined by Tech. Specs. will not be reduced. No unreviewed safety question is created by this change.

TN/1/A/1178/00/01A, Initial Issue

This procedure will provide work activities necessary for the replacement of the existing Unit 1 Events Recorder System per Nuclear Station Modification (NSM) CN-11178, Rev. 00.

The cables being pulled per this procedure are Non-Safety cables. These cables will not be terminated per this procedure. These cables will be routed such that the separation requirements as stated in the FSAR are maintained. The requirement for firewatches to be established when fire boundaries are breached is written in this procedure. This procedure will involve the penetration of a Control Room Ventilation System (VC) firestop. Because of this, steps, notes, and warnings have been incorporated in this procedure to ensure VC operability during the implementation of this procedure.

This procedure will also direct work activities for the cable terminations and wiring necessary for the replacement of the existing system as well as the installation of all components to the new Dranetz system. The terminations and wiring will include 27 new Operator Aid Computer (OAC) points which were originally wired via the Events Recorder. For these work activities to take place, the entire Unit 1 Events Recorder will be removed for service. This will occur while Unit 1 is on line.

The Events Recorder is not Tech. Spec. related. The bases for this system is to perform monitoring of plant systems, and it is also used for post trip analysis. The bases for this system will not change as a result of this procedure. The new system will be installed and tested before it is returned to service. These tests will ensure that the new system will perform all of its intended functions per the design bases. No other systems will be affected by this procedure, and as a result no FSAR changes will be created

from the implementation of this procedure. Accordingly, this procedure will not create any unreviewed safety questions.

PT/1/A/4200/26, Change #35

The purpose of this test is to satisfy the requirements of Section XI, Subsection IWV of ASME Boiler and Pressure Vessel Code with regard to the measurement of valve stroke time, valve operability and valve position indicator verification as required by Catawba IWV Submittal.

After the system is properly aligned, the valve is moved to its "Required Initial Test Position." The valve is then stroked, and the stroke time is recorded. The system is returned to its normal alignment as necessary. Once every two years, valve operation will be verified to agree with the remote position indicators and Operator Aid Computer (OAC) position indication.

Originally this test created a possible situation for the Containment Spray (NS) pumps to start without a suction source. By defeating NS pumps' auto start circuitry, this change allows testing with breakers racked in, prevents pump start without suction source, and reduces the possibility of equipment malfunction.

During this test, one train of NS will remain operable. The opposite train will be inoperable for a period of time well within the action time allowed by Tech. Specs.; therefore, the margin of safety as defined of Tech. Specs. will not be reduced. No unreviewed safety questions are created by this change.

PT/2/A/4200/26, Change #22

The purpose of this test is to satisfy the requirements of Section XI, Subsection IWV of ASME Boiler and Pressure Vessel Code with regard to the measurement of valve stroke time, valve operability and valve position indicator verification as required by Catawba IWV Submittal.

After the system is properly aligned, the valve is moved to its "Required Initial Test Position". The valve is then stroked, and the stroke time is recorded. The system is returned to its normal alignment as necessary. Once every two years, valve operation will be verified to agree with the remote position indicators and OAC position indication.

Originally the breakers were racked out to prevent the NS pumps from starting without a suction source. This change will allow testing with the NS pump Breakers racked in and prevents NS Pump start without a suction source by defeating auto start circuitry. This change will also increase the availability of the system by limiting its inoperable status to approximately one (1) hour as compared to approximately four (4) to five (5) hours with breakers racked out.

During this test, one train of the NS will remain operable. The opposite train will be inoperable for a period of time well within the action time allowed by Tech. Spec.; therefore, the margin of safety as defined by Tech.

Specs. will not be reduced. No unreviewed safety questions are created by this change.

OP/0/B/6250/10, Retype #6, Changes 0 to 14 Incorporated

This change incorporates the use of morpholine as a chemical additive in the secondary system of the plant. The addition of morpholine is considered beneficial in the control of erosion/corrosion. The following is a detailed technical evaluation of the use of morpholine.

Technical Summary for Morpholine Addition to Secondary Cycle at Catawba Nuclear Station

Introduction

The main purpose of adding morpholine to the Secondary Coolant is to reduce the iron transport to the steam generators. An additional benefit is that a reduction of corrosion at any point in the system results in an extended life for that system. The current corrosion inhibition function in the Catawba secondary system is performed by All Volatile Treatment (AVT) Chemistry. This consists mainly of additions of hydrazine (which decomposes to ammonia) and occasional additions of ammonia to maintain a pH that effectively minimizes corrosion of the condensate/feedwater piping and components.

Ammonia, however, favors the vapor phase during the condensation process, and, therefore, the liquid portion of certain two phase flow areas of the system is at a less than optimal pH to limit corrosion. This has been demonstrated by extensive chemistry studies of corrosion product transport around the system.

Morpholine is also a volatile amine, but it partitions about evenly between the liquid and vapor phases and can therefore raise the pH in two phase flow areas of the system. These minor changes in pH can significantly decrease the rate of corrosion in the affected areas, which in turn results in a corresponding decrease in iron transport to the steam generators.

Until 1985, only a very few US utilities used morpholine in their Pressurized Water Reactor (PWR) plants. Since then, however, it has been more widely recognized and accepted by both Nuclear Steam Supply System (NSSS) vendors and the utilities for its potential to decrease buildup of iron oxides in steam generators. At the beginning of 1990, approximately 25 of 73 US PWRs were on combined AVT/morpholine chemistry, including both Oconee and McGuire. The use of morpholine was widespread enough for it to be incorporated into the most recent revision of the EPRI PWR Secondary Water Chemistry Guidelines as an alternative to conventional AVT Chemistry.

At the present time, morpholine addition is considered a test or experiment not described in the Catawba FSAR; however, it will be incorporated into the next revision. Therefore, this evaluation covers affected sections of the FSAR, Technical Specifications, and issues related to non-metallic materials compatibility.

Technical Evaluation

The following material reviews all changes and areas of concern for Catawba Chemistry to adopt morpholine addition to the secondary coolant.

1.0 FSAR Sections Reviewed for Applicability

1.1 FSAR Section 10.3, "Main Steam Supply System," Subpart 10.3.5.2 "Secondary Side Water Chemistry". A short discussion of the purpose of maintaining water purity, the treatment used, the monitoring requirements, and controlling chemistry is presented. Each of these will be considered separately as it applies morpholine addition.

1.1.1 Maintaining Water Purity (10.3.5.2)

The reasons given for maintaining water purity within specified limits are to minimize corrosion and minimize fouling of steam generator surfaces. The addition of morpholine as described is to retard these processes. This is considered to be in agreement with the stated purpose and, therefore, does not represent a significant deviation to the FSAR.

1.1.2 Treatment (10.3.5.2.1)

The AVT treatment is described as the addition of hydrazine for oxygen scavenging and the addition of ammonia for pH control. The addition of morpholine has no effect on the ability of hydrazine to scavenge oxygen. The addition of morpholine is expected to slightly raise the pH as maintained by ammonia in two phase portions of the system. However, morpholine is not expected to change the pH of steam and/or feedwater significantly. Morpholine is therefore being used in conjunction with ammonia (or hydrazine which decomposes to ammonia) to maintain a more desirable system pH. Powdered resin demineralizers are mentioned as used for condensate polishing and will be discussed under 10.4.6, "Condensate Cleanup System".

1.1.3 Monitoring (10.3.5.2.2)

The sampling and analysis program for feedwater, condensate, and steam generators, as related to current operating guidelines, is referenced to vendor recommendations and the current revision of the EPRI PWR Secondary Water Chemistry Guidelines. Many Westinghouse plants are using morpholine/AVT treatment, and its use is supported by Westinghouse. The effects of morpholine addition on pH and also conductivity are covered in detail in the EPRI Guidelines.

1.1.4 Controlling Chemistry (10.3.5.2.3)

The use of blowdown demineralizers, polishers, makeup water, and chemical additives to maintain chemistry limits are addressed here. The blowdown demineralizers will be discussed under 10.4.8, "Steam Generator Blowdown System." Since detailed description of chemistry control is left to Plant Operating Procedures, addition of morpholine is considered to be appropriately handled the same way.

1.2 FSAR Section 10.4, "Other Features of Steam and Power Conversion System", Subparts 10.4.1, 10.4.6, 10.4.7, 10.4.8, and 10.4.9 are reviewed separately as they pertain to morpholine addition.

1.2.1 10.4.1, "Main Condenser", subpart 10.4.1.4, "Test and Inspections", lists the monitoring requirements for condensate leaving the hotwell as it relates to condenser cooling water leakage. No effects are expected that will minimize the ability to identify condenser leaks. The measurement of the most sensitive parameter, sodium, will not be affected by the presence of morpholine at the concentration of condensed main steam.

1.2.2 10.4.6, "Condensate Cleanup System (CCS)", describes the mode of operation of the system and provides the design bases, which is to remove dissolved and suspended impurities, radioactivity in the event of a primary to secondary leak, and additional impurities in the event of a condensate leak. The individual polishers will be allowed to saturate on morpholine. Utility experience has shown the polisher's ability to remove these contaminants will not be significantly impacted by the presence of morpholine.

1.2.3 10.4.7, "Condensate and Feedwater Systems", describes the functions of these two systems to provide water to the steam generators while maintaining proper water inventories throughout the cycle. The addition of morpholine will not affect these systems' ability to perform their design function in any manner.

1.2.4 10.4.8, "Steam Generator Blowdown System", is designed to maintain proper shell side water chemistry by removing non-volatile materials and provide a purification and recovery system for the blowdown. The mixed bed demineralizers described for purification will be allowed to saturate on morpholine, a process that takes some time. Utility experience has shown the demineralizers' ability to remove contaminants will not be significantly impacted by the presence of morpholine.

1.2.5 10.4.9, "Auxiliary Feedwater System", is designed to provide sufficient feedwater supply to the steam generators in the event of loss of feedwater/condensate.

The preferred quality of water for this system is condensate grade, but it can be aligned to the Standby Nuclear Service Water Pond. Morpholine addition in no way affects the auxiliary feedwater system's ability to perform its design function and may actually provide some protection from corrosion processes to the generators in the event Service Water is used.

1.3 Summary

It can be concluded from this information that Secondary Cycle pH control additives do not play a significant role in the accident and malfunction scenarios outlined in the FSAR. They also do not create additional scenarios of this type not considered in the FSAR. Their purpose is to minimize the corrosion of metallic surfaces and, therefore, extend the life and improve the reliability of the components they come in contact with.

2.0 Technical Specifications Reviewed for Applicability

3/4.7 Plant Systems

The addition of morpholine will not affect the operability of any of the systems listed in this section.

3/4.11 Radioactive Effluents

Morpholine is not a radioactive material and, therefore, does not affect this section.

6.8.4.c Procedures and Programs

The pertinent section of Technical Specifications dealing with Secondary Chemistry is subpart 6.8.4 (c). This summarizes the program required for monitoring secondary water chemistry to inhibit steam generator tube degradation. Since morpholine addition contributes specifically to this purpose, no change to this Tech. Spec. is needed.

Appendix B: Environmental Protection Plan

The use of morpholine in the secondary system will result in the discharge of morpholine through the Conventional Waste (WC) system and its associated National Pollution Discharge Elimination System (NPDES) discharge point. Administrative limits of 10 ppm have been previously approved by South Carolina Department of Health and Environmental Control (SCDHEC) for Catawba through either the Conventional Wastewater (WC) or Low Pressure Service Water (RL) systems. Therefore, it is determined that the use of morpholine in the secondary system will be consistent with the guidance and limitations provided by SCDHEC.

3.0 Secondary Non-Metallic Materials Compatibility

In general, non-metallic materials used in the Secondary System are in the form of gaskets, diaphragms, O-rings, seals, etc. and are made of such materials as asbestos, teflon, viton, EPDN, and rubber (Buna-N, Butyl, black, red). These are all common materials in use in any nuclear plant on similar equipment. The concentration of morpholine to be used in the feedwater/condensate system is not considered to be a concern to the integrity of these materials based on industry experience and testing programs. Only the chemical addition system will see concentrations of morpholine that might be considered harmful. Therefore, a survey was conducted of non-metallic materials in use in the chemical feed pumps, tank, and various instrumentation associated with this system. The chemical feed pumps have teflon diaphragms, and the tank and associated piping are all stainless steel. Instrumentation in this system consists of pressure and flow gauges not affected by 1-5% morpholine.

For these reasons, these changes do not create an unreviewed safety question.

PT/0/A/4400/10, Retype, Changes 0 to 0 Incorporated

This procedure performs a spring test to determine the opening force of check valves that cannot be tested normally by a full flow test. The valves require disassembly using the appropriate disassembly procedure prior to performing this procedure. The following changes have been incorporated into this re-type of this procedure.

1) Valves 1&2 NI 248, 249, 250, 251, 252, and 253 were deleted from this procedure. These valves were part of the Upper Head Injection system prior to its removal.

2) Valves 1&2 NI 125, 126, 129, 134, 175, 176, 180, and 181 were added to this procedure. These valves have been added to the Inservice Test (IST) program. Baseline inspections were performed on valves 1&2 NI 125 and 175. From the baseline test, the maximum allowable force for these valves has been determined to be 12 pounds.

The appropriate sections of the FSAR and TECH. SPEC. have been reviewed and have been determined not to be affected. Since these changes incorporate recent revisions of the IST program, this procedure complies with the present testing requirements. The health and safety of the public has not been affected. No unreviewed safety questions exist.

PT/1/A/4150/26, Retype, Changes 0 to 3 Incorporated

This 10CFR50.59 Evaluation is for a major rewrite of PT/1/A/4150/26. Since major changes to the valves tested under this procedure have occurred, a full evaluation has been performed.

The following changes have been made in this rewrite.

1. Valves NI-248, 249, 250, 251, 252, and 253 have been removed from this procedure. These valves were removed from service under NSM CN-10910, Rev. 0, by deletion of the Upper Head Injection piping. Further testing is not required.

2. Valves NI-125, 126, 129, 134, 175, 176, 180, and 181 have been added to this procedure. These valves were added to the Inservice Testing Program per Revision 17 (Unit 1) of the Test Manual.

3. Add last test date of 1/90 on refueling outage #4 for valves CA-172, NI-93, NI-94, NI-125, NI-175, NS-21, NS-33, NS-99, SA-6, KD-6, and KD-21.

All changes comply with the latest revision of the Catawba Nuclear Station Pump and Valve Inservice Testing Manual. The FSAR and Tech. Specs. have not been affected. No unreviewed safety questions exist.

PT/2/4150/26, Retype, Changes 0 to 1 Incorporated

This 10CFR50.59 Evaluation is for a major rewrite of PT/2/A/4150/26. Since major changes to the valves tested under this procedure have occurred, a full evaluation has been performed.

The following changes have been made in the rewrite.

1. Valves NI-125, 126, 129, 134, 175, 176, 180, and 181 have been added to this procedure. These valves were added to the Inservice Testing Program per Revision 8 (Unit 2) of the Test Manual.

2. Add last test date of 6/90 on refueling outage #3 for valves CA-171, NI-81, NI-82, NI-125, NI-175, NS-4, NS-30, NS-98, SA-6, KD-6, and KD-21.

All changes comply with the latest revision of the Catawba Nuclear Station Pump and Valve Inservice Testing Manual. The FSAR and Tech. Specs. have not been affected. No unreviewed safety questions exist.

PT/2/A/4150/13D, Change #5

The opening of Tempering Flow on Steam Generator (S/G) D will introduce an unaccounted for flow to the system. Thus, Secondary Power will decrease by approximately 3.1%. Due to the reduction in Main Feedwater (CF) Venturi Flow, which is an input to the Secondary Power Calculation, power will be established at 96% prior to the test. Since the Secondary Power Calculation will no longer be accurate, primary power will be used to control Reactor Power. This will ensure that the Tech. Spec. Rated Thermal Power is not exceeded. Therefore, the margin of safety will not be reduced.

The opening of Tempering Flow will not change the existing flow path to the Auxiliary Feedwater (CA) Nozzle on S/G D. Therefore, the probability or consequences of a malfunction of equipment due to this test will not increase. No new accident scenerios are created, and the probability or consequences of an accident previously evaluated will not increase. No unreviewed safety questions are created.

TN/1/A/1067/00/01A, Initial Issue

NSM CN-11067, Rev. 0, will provide a manual bypass of the P-14 safety signal (S/G Hi-Hi Level) to prevent nuisance feedwater isolations during modes 4, 5, and 6. This procedure provides guidelines for pulling cables, performing wiring changes, installing and wiring the switches on the Control Board 1MC3, performing Solid State Protection System (SSPS) cabinet changes, and performing a functional and retest for the modification.

No work will begin on this procedure until Unit One is in modes 5, 6, or No Mode. No system will be prevented from performing any function important to safety while this work is being performed.

During the implementation of this procedure, as a result of the isolations, various indicating lights, status lights, annunciators, and events recorder points will be out of service. None of these indications are required to be operable during the modes that we are implementing the procedure.

Also as a result of the isolations, the Refueling Water Storage Tank (FWST) Heater groups A and B will be inoperable. This will not be a problem since we will be implementing this procedure in March, April, and May.

This procedure also removes power from 1SSPSA and 1SSPSB. However, the Solid State Protection System is not required to be operable in modes 5, 6, and No mode. (Reference Technical Specifications 3.3.1 and 3.3.2.) The Solid State Protection System is required for Mode 4 and will be returned to service prior to Unit One entering Mode 4 up.

The isolations made for the implementation of this procedure will also affect valves 1VC008A, 2VC008A, and 1YC121B. Valves 1VC008A and 2VC008A will be red tagged closed for the implementation of Step 8.3.4.3. Valve 1YC121B will be red tagged closed for the implementation of Step 8.3.5.4. Red tagging these valves closed will not make either train of the Control Room Ventilation/Chilled Water Systems (VC/YC) inoperable.

1VC008A is the A Train pressure filter train cross-connect valve. When VC/YC Train A is put in service, this valve automatically closes for train separation, and the B Train cross-connect valve, 2VC007B, automatically opens. If VC/YC Train A is in service during the implementation of Step 8.3.4.3, red tagging 1VC008A closed will not be an operational concern since it's supposed to be closed when VC/YC Train A is in service. If Train B VC/YC is in service during the implementation of Step 8.3.4.3, red tagging 1VC008A closed will not be an operational concern since the B Train cross-connect valve will be closed also. The only time this valve is needed to open is when either pressure filter train carbon bed needs to be cooled. If this happens during the implementation of this step, Operations can manually open 1VC008A.

2VC008A is a B Train miniflow isolation valve. This valve has no automatic functions. This is a normally closed valve. Red tagging this valve closed for the implementation of Step 8.3.4.3 will not be an operational concern. It is only opened when Train B pressure filter train carbon bed needs to be cooled. If this happens during the implementation of Step 8.3 4.3, Operations can manually open 2VC008A.

1YC121B provides chilled water makeup for VC/YC Train B from the Makeup Demineralized Water System (YM). YM is a non-safety system and is not required to be available in an accident condition. 1YC121B receives a signal to close on a safety injection. Red tagging this valve c sed for the implementation of Step 8.3.5.4 will not be an operational concern since this valve's fail safe position is closed.

Accordingly, the implementation of this procedure will not increase the probability or consequences of an accident previously evaluated, or different than any already evaluated, in the FSAR. Nor will the implementation of this procedure increase the probability or consequences of an equipment malfunction previously evaluated, or different than any already evaluated, in the FSAR. The margin of safety defined in the bases of the Technical Specifications is unaffected and no unreviewed safety questions exists.

TN/1/A/1005/02/03A, Initial Issue

Nuclear Station Modification (NSM) CN-1 05/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the rework of one support on Math Model KCH. This math model includes supports only on the Component Cooling (KC) system.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model KCH has been qualified for the present support/restraint configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration with the support being partially dismantled has not been analyzed. For this reason, Design has determined this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified provided the modifications to the existing supports have been completed. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of modified supports provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/04A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the rework of one support on Math Model KCG. This math model includes supports only on the Component Cooling (KC) system.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model KCG has been qualified for the present support/restraint configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration with one support being partially dismantled has not been analyzed. For this reason, Design has determined this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified, provided the modifications to the existing supports have been completed. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of modified supports provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/06A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the rework of one support on Math Model KCZ. This math model includes supports only on the Component Cooling (KC) system.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model KCZ has been qualified for the present support/restraint configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration with the support being partially dismantled has not been analyzed. For this reason, Design has determined this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected

system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified provided the modifications to the existing supports have been completed. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of modified supports provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/10A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the removal of snubbers deleted from Math Model ASF. This math model includes supports on the Auxiliary Steam (AS), Chemical and Volume Control (NV), and Liquid Radwaste (WL) systems. These supports will either be deleted from the system or revised to a different configuration.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model ASF has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified under any combination of snubbers removed provided the modifications to the existing supports have been completed. This procedure is written to complete all modifications before deleting any supports. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of modified supports provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Based on this discussion, there are no unreviewed safety questions associated with this procedure.

TN/1/A/1005/02/13A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the rework of one support on Math Model NDO. This math model includes supports only on the Residual Heat Removal (ND) system.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model NDO has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration with the support being partially dismantled has not been analyzed. For this reason, Design has determined this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified provided the modifications to the existing supports have been completed. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of modified supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/14A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the removal of snubbers deleted from Math Model NBE. This math model includes supports on the Boron Recycle (NB) system. These supports will be deleted from the system.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model NBE has been qualified for the present support/restraint configuration. It has also been qualified for the support/restraint (S/R) configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours

allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified under any combination of snubbers removed. However, if the support deletions cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of deleted supports provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/15A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the removal of snubbers deleted from Math Model NDD. This math model includes supports on the Hydrogen Blanket (GB) and Residual Heat Removal (ND) systems. These supports will either be deleted from the system or revised to a different configuration.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model NDD has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed, because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified under any combination of snubbers removed provided the modifications to the existing supports have been completed. This procedure is written to complete all modifications before deleting any supports. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of modified supports provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/17A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the removal of snubbers deleted from Math Model NVD. This math model includes supports on the Chemical and Volume Control (NV) system. These supports will either be deleted from the system or revised to a different configuration.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model NVD has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed, because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified under any combination of snubbers removed, provided the modifications to the existing supports have been completed. This procedure is written to complete all modifications before deleting any supports. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of modified supports provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Based on this discussion, there are no unreviewed safety questions associated with this procedure.

TN/1/A/1005/02/19A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the removal of snubbers deleted from Math Model KDA. This math model includes supports on the Diesel Generator Engine Cooling Water (KD) system. These supports will be deleted from the system.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model KDA has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed,

because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified under any combination of snubbers removed. This procedure contains only deleted supports. However, if the support deletions cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of deleted supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/22A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the removal of snubbers deleted from Math Model CFA. This math model includes supports on the Main Feedwater (CF) system. These supports will either be deleted from the system or revised to a different configuration.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model CFA has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified under any combination of snubbers removed provided the modifications to the existing supports have been completed. This procedure is written to complete all modifications before deleting any supports. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of modified supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure

will provide the necessary controls to ensure compliance with all technical specification requirements.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/24A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the removal of snubbers deleted from Math Model NM-03. This math model includes supports on the Nuclear Sampling (NM) system. These supports will either be deleted from the system or revised to a different configuration.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model NM-03 has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed, because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified under any combination of snubbers removed, provided the modifications to the existing supports have been completed. This procedure is written to complete all modifications before deleting any supports. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of modified supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/25A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the removal of snubbers deleted from Math Model NV-04 This math model includes supports on the Chemical and Volume Control (NV) system. These supports will either be deleted from the system or revised to a different configuration.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model NV-04 has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed, because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified under any combination of snubbers removed, provided the modifications to the existing supports have been completed. This procedure is written to complete all modifications before deleting any supports. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of modified supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/26A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the removal of snubbers deleted from Math Model NV-01. This math model includes supports on the Chemical and Volume Control (NV) system. These supports will either be deleted from the system, or revised to a different configuration.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model NV-01 has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed, because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified under any combination of snubbers removed, provided the

modifications to the existing supports have been completed. This procedure is written to complete all modifications before deleting any supports. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of modified supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Based on this discussion, there are no unreviewed safety questions associated with the implerentation of this procedure.

TN/1/A/1005/02/27A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the removal of snubbers deleted from Math Model NV-13. This math model includes supports on the Chemical and Volume Control (NV) and Liquid Radwaste (WL) systems. These supports will either be deleted from the system or revised to a different configuration.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model NV-13 has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed, because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified under any combination of snubbers removed, provided the modifications to the existing supports have been completed. This procedure is written to complete all modifications before deleting any supports. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of modified supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/28A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the removal of snubbers deleted from Math Model NM-04. This math model includes supports on the Nuclear Sampling (NM) system. These supports will either be deleted from the system or revised to a different configuration.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model NM-04 has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed, because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified under any combination of snubbers removed, provided the modifications to the existing supports have been completed. This procedure is written to complete all modifications before deleting any supports. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of modified supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/30A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the rework of one support on Math Model KCW. This math model includes supports only on the Component Cooling (KC) system.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model KCW has been qualified for the present support/restraint configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration with the support being partially dismantled has not been analyzed. For this reason, Design has

determined this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified, provided the modifications to the existing supports have been completed. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of modified supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/31A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the rework of one support on Math Model KCY. This math model includes supports only on the Component Cooling (KC) system.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model KCY has been qualified for the present support/restraint configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration with the support being partially dismantled has not been analyzed. For this reason, Design has determined this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified, provided the modifications to the existing supports have been completed. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of modified supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/3099/CE/01A, Initial Issue

In order to comply with the ASME Code Section XI Inservice Inspection requirements, a radiographic examination must be performed on the weld (1SM34-7) located between Steam Generator (S/G) 1A and the 32" diameter main steam (SM) piping. In order to perform this radiography efficiently, a hole is needed in the SM system piping to allow access inside the piping. Exempt Change CE-3099 was originated to provide the access hole for the radiographic examination of the above weld. After the examination is complete, the hole will be plugged. The purpose of this procedure is to provide guidance for the drilling of the hole and the installation of the half coupling and plug.

The Operations Group will coordinate the isolations necessary to implement this procedure. S/G 1A and its associated main steam line will be out of service during the installation phase of this procedure. The drilling of the hole and the installation of the half coupling and plug will be performed during Modes 5, 6, and no mode. The Operations Group will ensure containment integrity is maintained while the affected main steam line is open to containment.

Testing for CE-3099 will be performed in accordance with the Post Modification Testing program at the station. Since the hole is one inch in diameter, a pressure test is not required per ASME Section XI exemptions; however, a visual inspection for leaks will be performed in Mode 3 with the affected SM piping at normal system temperature and pressure. This inspection will not affect the function or operation of the SM system. System insolations or abnormal alignments are not required.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/3213/CE/01A, Initial Issue

In order to comply with the ASME Code Section XI Inservice Inspection requirements, a radiographic examination must be performed on the weld (1SM-1A-F) located on the 32" diameter main steam (SM) piping from Steam Generator (S/G) 1A. In order to perform this radiography efficiently, a hole is needed in the SM system piping to allow access inside the piping. Exempt Change CE-3213 was originated to provide the access hole for the radiographic examination of the above weld. After the examination is complete, the hole will be plugged. The purpose of this procedure is to provide guidance for the drilling of the hole and the installation of the half coupling and plug.

The Operations Group will coordinate the isolations necessary to implement this procedure. S/G 1A and its associated main steam line will be out of service during the installation phase of this procedure. The drilling of the hole and the installation of the half coupling and plug will be performed during Modes 5, 6, and no mode. The Operations Group will ensure containment integrity is maintained while the affected main steam line is open to containment.

Testing for CE-3213 will be performed in accordance with the Post Modification Testing program at the station. Since the hole is one inch in diameter, a

pressure test is not required per ASME Section XI exemptions; however, a visual inspection for leaks will be performed in Mode 3 with the affected SM piping at normal system temperature and pressure. This inspectic will not affect the function or operation of the SM system. System isolations or abnormal alignments are not requ d.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

MP/0/A/7400/70, Initial Issue

This is a new procedure that was developed to provide a method for the controlled removal and replacement of a maintenance tool (baring device) that is normally stored on the engine. The safety circuitry that allows engagement or removal of the maintenance tool (baring device), also prevents normal air operation of the engine.

This procedure assures that the maintenance tool (baring device) safety circuitry is complete and that the air system is functioning as designed whether or not the maintenance tool (baring device) is installed.

The maintenance tool (baring device) has no safety related function. The maintenance tool (baring device) assembly is not mounted on the engine, but is mounted on the floor. Removal of the maintenance tool from its base assembly, which is mounted in the floor, greatly reduces the weight on that base assembly. It further LOWERS the center of gravity from the maintenance tool to the floor mounted base. Installation of the spacer onto the base assembly is a very insignificant weight addition, but in fact, much, much less than the maintenance tool itself. The floor mounted base assembly with the spacer installed would NOT effect the engine or the generator in a seismic event. The parts that hold the pin in position are not loaded, except to resist the tiny spring in the switch, as when the maintenance tool is installed.

The procedure requires Post Maintenance Testing (verifying the engine will air roll) to assure proper system operation immediately after tool removal AND after tool replacement.

The procedure assures equipment availability, reliability, and operability through well thought out and evaluated specified steps. The procedure does NOT create an unreviewed safety question.

MP/0/A/7200/04, Change #2

This procedure has been revised; the changes to the procedure are as follows:

Section 11.0	Step 11.4.11.1 was changed to avoid running the turbine at speeds less than 2000 RPM. Speeds below 2000 RPM can cause internal damage to the pump.
Section 11.0	Step 11.5, a "Caution" was added to clarify when the overspeed trip device can be operated.

Section 11.0	Step 11.5.4 was revised to delete the MES holdpoint.
Enclosure 13.1	Was revised to delete appropriate sign-offs.
Title	The title was changed to include the words "Overspeed Testing" to give a better description of the content of the procedure.

Tech. Spec. 3/4.7.1 could be affected by this procedure. Operations has the responsibility and the procedures for compliance with this Tech. Spec. Maintenance will be performed on the governor when Tech. Specs. allow, per Operation's procedures. This revision will clarify and assure that maintenance activities will return the governor to as-designed conditions.

The changes made by this revision have been reviewed against approved vendor manuals, design documents, and station procedures to ensure that the corrective maintenance controlled by this procedure will return the governor to as-built/as-designed condition. These actions will ensure the governor's compliance with FSAR accident analysis. Since the governor will be returned to as-designed conditions, the possibility, consequences, or probability of a malfunction will be reduced.

Therefore, no unreviewed safety questions exist.

TN/1/A/1188/00/01A Initial Issue

Nuclear Station Modification (NSM) CN-11188, Rev. 0, will install the Inadequate Core Cooling Monitor-86 (ICCM-86) System Software Update PROMS V8.2 and Remote Display PROMS V.1.1 provided by Westinghouse in Field Change Notice (FCN) DCPO-40513A. This procedure will provide guidelines for installing the FCN, performing a functional verification of the FCN, and completing the Channel Calibrations for the Reactor Vessel Level Indication System (RVLIS).

The purpose of the Inadequate Core Cooling Monitor-86 (ICCM-86) System is to provide post-accident information of reactor core and reactor vessel conditions which would indicate the approach of inadequate core cooling conditions. The ICCM specifically monitors, processes, and displays information on the following parameters: incore thermocouple temperature, subcooling margin, and reactor vessel water level. According to Technical Specification 3.3.3.6, this instrumentation is required only in modes 1, 2, and 3.

No work will begin on this procedure until Unit One's reactor vessel head is removed (mode 6 entry). All modification work will be completed in modes 6 and no mode. Channel-Cals for RVLIS will be completed prior to mode 3 up. No system will be prevented from performing any function important to safety while this work is being performed. No unreviewed safety questions exist.

TN/1/A/0397/CE/01A, Initial Issue

This procedure relocates pushbuttons for valves 1CA54B and 1CA66B located on the Auxiliary Feedwater (CA) Pump Turbine Control Panel 1AFWPTCP.

Work activities within this procedure will require that the control functions for valves 1CA54B, 1CA66B, and 1CA85B be disabled. Disabling these control functions will require Unit 1 to be in Modes 4, 5, 6, or No Mode. The affected pushbuttons will be relocated and returned to service per TN/1/A/0397/CE/01A. Upon completion of the implementation of this procedure, the pushbutton for valves 1CA54B and 1CA66B will be stroked from 1AFWPTCP and the Control Room to verify proper operation. No work will be performed on the circuitry for valve 1CA85B. This valve was isolated only to prevent the swapping of its controls to 1AFWPTCP while the testing for valves 1CA54B and 1CA66B is being performed. No other systems, structure or components will be affected by the implementation of this procedure. As a result, the implementation of this procedure will not create an unreviewed safety question.

MP/0/A/7150/67, Retype, Changes 0 to 11 Incorporated

This procedure is for performing latching and unlatching operations on the Control Rod Drive Mechanism (CRDM) drive shafts. Instruction manuals CNM-1201.00-0030 and CNM-1201.13-038 provided technical information for the development of this procedure. Plant specific information was also included. This evaluation is for changes made during the procedure review following Catawba Unit 2 End of Cycle 3 outage. The Catawba FSAR and Technical Specifications have been reviewed and are not affected by this revision. The probability of an accident or a malfunction of equipment previously addressed will not be increased nor will any unreviewed safety questions be involved.

The following changes were made.

* Step 11.13.31.7 — Added to repeat steps if weight is reduced below tool weight minus 140 lbs.

* Step 11.14.28 — Added step to perform visual verification of drive rod locking screw position to insure drive shafts are locked.

* Step 13.0 — Added enclosure 13.5 Control Rod Drive Shaft.

* Revised Enclosure 13.1 Data Sheets for the above changes.

MP/1/A/7150/42, Retype, Changes 0 to 13 Incorporated

This procedure is for performing removal and replacement activities required on the Reactor Vessel Head during refueling outages.

This evaluation is for changes made during the procedure review following the Catawba Unit 1 End-of-Cycle 4 outage. These changes were required to clarify steps for better understanding and to add manufacturer's information on new equipment. See procedure changes below:

* Step 6.6 — Deleted this step.

* Step 8.1 — Moved special tool list to an enclosure.

*	Step 8.2	Added visual inspection for all lifting devices.
*	Step 9.1	Change stud elongation to (+.008"/-.002").
*	Step 10.2	Changed must to should.
*	Step 11.2	Revised temperature reference from Nil-Ductility Transition Temperature (NDTT) to minimum head flange temperature for detension/tension as specified in OP/1/A/6700/01.
*	Step 11.3.8	Deleted this step. Head stand shielding not required.
*	Step 11.4.2	Revised NDTT temperature reference as in 11.2.
*	Step 11.4.5	Added note 6.
*	Step 11.4.9	Changed spherical washer to top of the closure stud.
*	Step 11.4.22	Added compensating weights.
*	Step 11.4.24	Added step for cleaning stud hole plug sealing surface.
*	Step 11.4.25	Deleted this step.
*	Step 11.4.26	Added step for cleaning stud holes for installation of guide pin sleeves.
*	Step 11.4.33	Changed install to inspect.
*	Step 11.4.34	Added note.
*	Step 11.4.38	Changed to 11.4.37.4.
*	Step 11.4.38	Changed to install head ladder.
*	Step 11.4.50	Added note 4.
*	Step 11.4.56	Added step for inside shroud inspection.
*	Step 11.5.23	Added note 2.
*	Step 11.5.26.2	Deleted note 2.
*	Step 11.5.30	Revised NDTT temperature as in step 11.2.
*	Step 11.5.58	Note 2 Revised elong tion (+.008/-.002).
*	Step 11.5.59.3	Revised pump pressu e to 6600 psig.
*	Step 11.5.60.2	Revised elongation (+.008/-.002).
*	Step 11.5.61.6	Deleted this step.

* Step 11.5.61.10 Revised pump pressure to 6600 psig.

* Step 11.5.61.14 Revised elongation (+.008/.002).

* Step 13.0 Added 13.7 enclosure, Nozzle Port Cover Location.

* Enclosure 13.1 Revised for the changes made.

This procedure will be used to maintain the Reactor Vessel Head in its original design requirements and specifications. The Catawba FSAR and Technical Specifications have been reviewed and are not affected by this procedure change. The probability of an accident or a malfunction previously addressed will not be increased nor will any unreviewed safety question be involved.

MP/0/A/7300/14, Initial Issue

This procedure has been written to provide generic instructions for pillow block bearing inspection and lubrication on safety related fans.

The procedure has been written in a manner to maintain the equipment it is used on in its as-designed/as-built condition. This is ensured by having a step in the procedure to obtain the approved vendor technical manual for the specific piece of equipment before any work can begin. The approved technical manual is then referenced in several places in the procedure to establish the correct as-designed/as-built condition of the equipment.

It has been determined that when lock washers are present, bolts should not be torqued but tightened securely. The results of this determination are combined into the notes for steps 11.3.8 and 11.4.8 of the procedure. Realizing that there will be questions on this topic, the Technician is referred back to the Maintenance Engineering Services (MES) Engineer to determine the correct resolution. Design will be contacted as needed to resolve problems.

This procedure has been written based on approved vendor manuals, design documents, and station procedures to ensure that the actions controlled will return the fan bearings to their as-designed/as-built condition. Since the fan bearings will be returned to as-designed conditions, the possibility, consequences, or probability of a malfunction will be reduced. Therefore, no Unreviewed Safety Question exists.

PT/2/A/4200/09A, Change #74

This change was written so that only one Steam Generator (S/G) Power Operated Relief Valve (PORV) will be isolated during testing. Presently, all four S/G PORV block valves are closed, and all four PORVs are cycled during the test. Tech. Specs. require that three PORVs must be operable. A recent Tech. Spec. interpretation for Tech. Spec. 3.7.1.6 (approved 1/17/91) states that for the PORV to be operable, the PORV block valve must be OPERABLE and OPEN. This is to ensure that two PORVs are available to cool Reactor Coolant (NC) assuming a S/G tube rupture concurrent with a loss of offsite power.

This change will ensure that two PORVs are available for a cooldown to Residual Heat Removal initiation temperature in the event of a S/G tube rupture concurrent with a loss of offsite power and failure of a Diesel Generator. Only one S/G PORV block valve will be closed by the procedure to ensure that two PORVs are available to cooldown. No Unreviewed Safety Questions are created.

PT/1/A/4200/09A, Change #137

This change was written so that only one Steam Generator (S/G) Power Operated Relief Valve (PORV) will be isolated during testing. Presently, all four S/G PORV block valves are closed, and all four PORVs are cycled during the test. Tech. Specs. require that three PORVs must be operable. A recent Tech. Spec. interpretation for Tech. Spec. 3.7.1.6 (approved 1/17/91) states that for the PORV to be operable, the PORV block valve must be OPERABLE and OPEN. This is to ensure that two PORVs are available to cool Reactor Coolant (NC) assuming a S/G tube rupture concurrent with a loss of offsite power.

This change will ensure that two PORVs are available for a cooldown to Residual Heat Removal initiation temperature in the event of a S/G tube rupture concurrent with a loss of offsite power and failure of a Diesel Generator. Only one S/G PORV block valve will be closed by the procedure to ensure that two PORVs are available to cooldown. No Unreviewed Safety Questions are created.

PT/1/A/4200/09A, Change #136

This change was written to ensure that the normal letdown path is not isolated as a result of the Auxiliary Safeguards Test Cabinet Periodic Test. This is being done to prevent a possible thermal transient as a result of isolating letdown (reference Westinghouse report, VIL-W/90-40).

The change deletes an option which was in the procedure that has not been used in the last few years. The change will reduce the likelihood of a thermal transient. The change has no effect on the method in which this test has been performed within recent history. No Unreviewed Safety Questions are created by this change.

PT/2/A/4200/09A, Change #73

This change was written to ensure that the normal letdown path is not isolated as a result of the Auxiliary Safeguards Test Cabinet Periodic Test. This is being done to prevent a possible thermal transient as a result of isolating letdown (reference Westinghouse report, VIL-W 90-40).

The change deletes an option which was in the procedure that has not been used in the last few years. The change will reduce the likelihood of a thermal transient. The change has no effect on the method in which this test has been performed within recent history. This change does not create an Unreviewed Safety Question.

PT/1/A/4200/09, Retype, Changes 0 to 88 Incorporated

The Engineered Safety Features Actuation Periodic Test is performed in accordance with Section 7.3 of the Catawba FSAR and applicable Technical Specifications. Sections 12.1 and 12.4 of this test verify that ESF components actuate correctly to their safety position within their required time limits in response to a Safety Injection, Phase A Isolation, Phase B Isolation, and a Blackout. In addition, proper operation of the Diesel Generator (D/G) is verified. Sections 12.2 and 12.5 are essentially a repeat of 12.1 and 12.4, except there is no blackout of the essential switchgear, and the signals are generated usin~ the manual pushbuttons in the control room. Sections 12.3 and 12.6 verify proper D/G starting, load shedding, and load sequencing in response to a blackout. Correct valve movements, response times, and system responses are verified. Sections 12.7 and 12.8 verify that each cold leg accumulator valve opens when a simulated Reactor Coolant (NC) system pressure signal exceeds the P-11 setpoint and upon receipt of Safety Injection Signal. Sections 12.9 and 12.10 verify the proper operation and response time of the automatic swapover to containment recirculation upon a Safety Injection coincident with Lo-Lo Refueling Water Storage Tank (FWST) level. Section 12.11 verifies that the Main Steam isolation valves, Main Steam isolation bypass valves, and the Steam Generator (S/G) Power Operated Relief Valves (PORVs) close upon a manual Main Steam isolation.

In sections 12.1 through 12.6, the opposite train of the one under test will remain operable during the test. All pumps on the train under test are inoperable (due to alignment to recirculation flow path) except for the Nuclear Service Water (RN) and Component Cooling (KC) pumps (in section 12.2 and 12.5), both of which are aligned for normal operation. The RN and KC pumps are inoperable in sections 12.1 and 12.4 due to the D/G being inoperable. The opposite train is capable of fulfilling any required safety functions. The discharge of the Chemical and Volume Control (NV) pump and the Safety Injection (NI) pump on the train under test are isolated from the Reactor Coolant System by two isolation valves to provide low temperature overpressure protection as required by Technical Specifications. In addition, as a precaution to avoid inadvertent actuation of an NC PORV, the prerequisites to the test specify that the NC system cannot be water solid. Tech. Spec. 3.1.2.1 requires that one boron injection flow path be operable. One boron injection flowpath will remain operable since the centrifugal charging pump on the opposite train of the one being tested is not rendered inoperable by this test. Radiation Monitors (EMFs) 38, 39, and 40, which provide interlocks to terminate containment release through Containment Purge Ventilation (VP) and Containment Air Release and Addition (VQ), will be rendered inoperable by the conduct of this test when the sample lines are isolated by a Phase A containment isolation signal; however, both VP and VQ operation will be terminated upon receipt of the Phase A containment isolation signal. The Spray/Sprinkler system will become inoperable due to the closure of the containment isolation valve to the annulus sprinklers. The test should be completed well within the one hour in which action must be taken. If the test cannot be finished within one hour, the action can be easily satisfied. A Limit and Precaution has been added to the procedure stating that portions of the Fire Protection System (RF) will be inoperable due to valves 1RF389B, 1RF447B, and 1RF457B receiving a Phase A containment isolation signal. In sections 12.1 and 12.4, the D/G on the train under test is inoperable due to the accelerated sequence being defeated. If the D/G runs for an extended

period of time, that D/G will become inoperable due to fuel oil level dropping below the minimum allowed by Tech. Specs.; however, only one D/G is required to be operable during Modes 5 and 6. Assuming an initial fuel oil supply of 80,000 gallons (Operations generally maintains tank between 82,000 and 86,000 gallons), the D/G would have to be run for greater than 7 hours before becoming inoperable due to fuel oil supply. This test should not run the D/G for more than 2 hours. The procedure requires the D/G on the opposite train to be operable. Section 12.7 and 12.8 do not make the Cold Leg accumulators inoperable, since they are only required in Modes 1 through 3, and the test is done in Modes 4, 5, or 6. Sections 12.9 and 12.10 will each make the Residual Heat Removal (ND) train under test inoperable due to the suction piping being drained. These sections will be done when the unit is below Mode 4, when only one train of ND is required. Only one train may be tested at a time. Section 12.11 does not make Main Steam inoperable and is performed in Mode 5 when Main Steam Isolation is not required.

Several additional changes have been made on this reissue of the procedure which were not previously approved.

1) Steps have been added to remove power from valves 1NI9A (sections 12.1 and 12.2) and 1NI10B (sections 12.4 and 12.5) if the opposite train NV pump is running. These steps were added to prevent injection into the NC system.

2) The S/G PORV block valves were added to the lineups so that opening the PORVs for the test would have no effect.

3) Valves 1CF21 and 1CF25 were added to enclosures 13.18A and 13.19A. These valves have been closed by Operations in the past to prevent the S/Gs from filling during the test. They have been added to the procedure to document that they are closed and that they are returned to their "As Found" position.

4) Enclosures 13.3.9 and 13.6.9 were added to aid Operations in recovering from the blackout sections of the test. The enclosures are to assist in returning systems (components) to their pretest alignment.

5) Steps have been added to verify status lights and annunciators that are expected during the test. These steps were added so that the test coordinator would be aware of any problems that may arise during the setup in the process cabinets, and so that the control room operators would know to expect the alarms.

6) The drawing number for the head curve for NI Pump 1A was corrected on Enclosure 13.1.7 and in Section 2.2.2. This has no effect on the performance of the test other than to ensure that the correct curve is used.

7) The setpoints for the pressure switches listed in step 8.0.9 were deleted since the setpoints are part of the calibration procedures.

8) A note was added prior to steps defeating the accelerated sequence stating that the D/G is inoperable.

9) Section 12.1 has been changed so that power will be removed from 1NS29A and 1NS32A to ensure that containment is not sprayed. In Section 12.2, the containment is not sprayed. In Section 12.2, the Containment Spray (NS) pump will be racked to test, and 1NS29A and 1NS32A will be stroked to verify response times.

10) Section 12.4 has been changed so that power will be removed from 1NS12B and 1NS15B to ensure that containment is not sprayed. In Section 12.5, the NS pump will be racked to test, and 1NS12B and 1NS15B will be stroked to verify response times.

11) In sections 12.3 and 12.6, the accelerated sequence will no longer be defeated. Performing the test with the accelerated sequence will more closely simulate the loading the D/G would actually see during a Blackout.

12) On enclosures 13.3.1A and 13.6.1A, Auxiliary Feedwater (CA) discharge valves to the S/Gs were added to ensure the CA Pump does not discharge to the S/Gs.

13) On Enclosure 13.6, the modifications for Air Return Fan (ARF) 1A were changed so that the test switch is no longer used. A jumper will be used to allow the bypass dampers to open.

14) In sections 12.1 and 12.4, the step starting the D/G visicorder has been moved to immediately prior to actuation to conserve paper to ensure the roll of paper does not empty during the test.

15) A caution has been added to restart Control Area Ventilation/Chilled Water Systems (VC/YC) if needed to ensure that control room pressure is maintained.

16) Additional guidance has been added to 13.3.6 and 13.6.6 outlining what to do if a Lower Containment Ventilation Unit does not start during the test.

17) Steps were added to sections 12.7 and 12.8 to Defeat CA Auto-Start when test for the simulated P-11 signal is completed. The test resets the Auto-Start defeat

18) On Enclosures 13.11.3 and 13.11.4, the "required time" was changed to 5 seconds to reflect Tech. Spec. 3.6.3 and IWV required times.

19) Additional breakers were added to 13.13 and 13.15 to ensure all breakers are returned their original pre-test status.

20) The procedure now requires valves that are used for double isolation of pumps be white-tagged to ensure the valves are not opened which could lead to a Tech. Spec. violation.

21) A prerequisite was added to sections 12.1 through 12.6 to verify the NC system is filled prior to the test. This is done since one train of ND will be taken out of service by the test. (See Tech. Spec. 3.4.1.4.1 and 3.4.1.4.2)

22) Limit and Precaution 6.2 has been deleted. Containment pressure is normally maintained within these limits.

23) The reactor trip breakers and Control Rod Drive Mechanism feeders were deleted from enclosures 13.3.2 and 13.6.2. The breakers are not needed to satisfy any alignments or interlocks, and they receive no signal during a blackout.

Pumps which receive a start signal are aligned to assure that a minimum flow path is available. The procedure requires immediate verification of flow upon actuation. Testing is performed in accordance with the FSAR and Technical Specifications.

This test does not create an unreviewed safety question.

PT/1/A/4250/13A, Retype, Changes 0 to 2 Incorporated

The following is a list of previously unapproved changes to PT/1/A/4250/13A. Items 1 through 6 are required because the use of dLOG has been incorporated into this procedure.

1) Change the "References Needed to Perform Test" (Section 2) from "None" to "dLOG Computer Program Benchmark".

2) Changes to Section 5 (Test Equipment):

   a) Require the use of transmitters instead of gauges.

   b) Require the use of Resistance Temperature Detectors (RTDs) instead of digital thermometers.

   c) Add requirement of precision resistor for each transmitter used.

   d) Add requirement of HP Data Logger

3) Changes to Section 8 (Prerequisite System Conditions)

   a) Add substep in test instrument installation step to attach RTDs to the suction and discharge piping of the pump.

   b) Add Steps 8.10 through 8.14 (including Note prior to 8.14) for dLOG setup.

4) Change test equipment listed in Section 10 (Data Required) to reflect changes given above for Section 5.

5) Changes to Section 12 (Procedure)

   a) Add Notes prior to steps 12.7 and 12.12 allowing the Test Coordinator to N/A Enclosure 13.4 if dLOG will be used to perform the Total Dynamic Head (TDH) calculations.

   b) Change step 12.9 (test instrument removal) to reflect changes in test equipment as described above.

6) Changes to Section 13 (Enclosures)

   a) Add Enclosures 13.9 and 13.10 for dLOG Setup.

   b) Change Enclosure 13.1 to reflect changes in test equipment.

   c) Delete Note on Enclosure 13.1 allowing blanks to be N/A'd if test equipment not used (test equipment must be used with dLOG).

   d) Change Enclosure 13.4 to require only one reading instead of three because dLOG can be used to obtain one accurate average value for a five minute period.

The following is a list of changes associated with the deletion of the requirement to take data at "max flow", thus making 600 gpm the highest flow data point.

1) Change the range on the discharge pressure gauge from 2000 psig to 1800 psig.

2) Delete the need for a 0-800 inwd D/P gauge.

3) Delete Notes prior to steps 8.1.1 and 12.7, and delete former step 8.1.2, concerning installation of the 0-800 inwd gauge.

4) Add Note prior to step 12.7 to avoid over-ranging of instrumentation (since the 0-800 inwd D/P gauge will no longer be used). Also, change the acceptance range for the 600 gpm flow point from (575-625 gpm) to (575-610 gpm) to avoid instrument over-ranging.

The option of taking vibration data was also deleted in this procedure retype because such data is obtained during each IWP test. The following changes were made to reflect this:

1) Delete need for vibration probe in Sections 5 and 10 (Test Equipment and Data Required, respectively).

2) Change the last sentence of the Test Method Section (Section 9) to indicate that no vibration data is required, and to explain the option of performing manual calculations, or using dLOG.

3) Delete Vibration data Note in Enclosure 13.4.

The remainder of changes included in the retype are listed below. Many of them reflect an attempt to standardize the head curve procedures (as much as possible) between units. Others are changes required to ensure that successful completion of the procedure can be achieved with minimal discrepancies.

1) The following References were added to Section 2:

   a) OP/1/A/6250/02, Auxiliary Feedwater System
   b) OP/1/A/6450/18, CA Pump Pit CO2 System

2) Add Limits and Precautions 6.5 and 6.6 to make sure pump miniflow limits are observed, and pump bearing temperatures are monitored by the Control Room Operator.

3) Change wording of CO2 and Firewatch steps to be consistent with the IWP procedures. The wording used (or to be used) for these steps in the IWP procedures is consistent with Operations' recommendations, and was developed through the Cross-Disciplinary Review process. Also remove the Independent Verification (I.V.) requirement for Operations to take the CO2 system out of service and return it back to service. The I.V. is not required because all actions taken are covered under OP/1/A/6450/18, CA Pump Pit CO2 System.

4) Add phrase "Have Operations..." to the beginning of each step dealing with the valve alignment.

5) Add Note concerning CA system operability to follow the step which has Operations align valves to their "Required Initial Test Position". This change was originated through a step in the two-year procedure review process.

6) Add step following step 12.7.1 to have the flow throttled to approximately 400 gpm before the pump is shut off. This is to allow Operations to readily complete their Head and Valve Verification procedure at the completion of this test.

7) Add Note following step 12.16 to ensure that the appropriate documents (Flow Balance Procedure, and Unit 1 Data Book) are updated if the % pump calculation result differs from the previous such result.

8) Add step to ensure that the Control Room Senior Reactor Operator and the Control Room Operator are informed of the completion of this test.

9) Change process gauge numbers in Enclosure 13.4 to the correct numbers.

10) Valve 1CA87, CA Pump 1A Discharge to S/G Isol, was added to the valve alignment to be required in the open position. This will further ensure proper system configuration during the performance of this test.

The purpose of this procedure is to verify that Auxiliary Feedwater (CA) Pump 1A is operating within acceptable limits, and to provide a pump strength value for the CA Flow Balance Procedure (PT/1/A/4250/03E).

The changes associated with the incorporation of the dLOG Computer Application for data acquisition do not alter the Test Method used. They do, however, change the method that data is acquired by utilizing a benchmarked computer data acquisition system to take data and perform calculations. The instruments used still meet the required accuracies stated in the CA System Test Acceptance Criteria Data Sheet CNTC-1592-CA-V001-02 (Rev. 1).

Other changes included in this retype delete the requirements to take vibration data, and head data at flows greater than 600 gpm. Although pump vibration is a critical parameter which should be monitored carefully, the deletion of the requirement to obtain vibration data via this procedure will not affect our pump performance monitoring program. This data is obtained regularly per the quarterly IWP procedure, PT/1/A/4250/03A. Limiting the range of flow rates at which Total Dynamic Head data is acquired to less than or equal to 600 gpm will not degrade the effectiveness of our pump performance monitoring. This range amply covers the required flow rates of CA Pump 1A in the event of an accident.

These changes do not create an Unreviewed Safety Question

PT/1/A/4250/13B, Retype, Changes 0 to 1 Incorporated

The following is a list of previously unapproved changes to PT/1/A/4250/13B. Items 1 through 6 are required because the use of dLOG has been incorporated into this procedure.

1) Change the "References Needed to Perform Test" (Section 2) from "None" to "dLOG Computer Program Benchmark".

2) Changes to Section 5 (Test Equipment):

   a) Require the use of transmitters instead of gauges.

   b) Require the use of Resistance Temperature Detectors (RTD's) instead of digital thermometers.

   c) Add requirement of precision resistor for each transmitter used.

   d) Add requirement of HP Data Logger

3) Changes to Section 8 (Prerequisite System Conditions)

   a) Add substep in test instrument installation step to attach RTDs to the suction and discharge piping of the pump.

   b) Add Steps 8.10 through 8.14 (including Note prior to 8.14) for dLOG setup.

4) Change test equipment listed in Section 10 (Data Required) to reflect changes given above for Section 5.

5) Changes to Section 12 (Procedure)

   a) Add Notes prior to steps 12.7 and 12.12 allowing the Test Coordinator to N/A Enclosure 13.4 if dLOG will be used to perform the TDH calculations.

   b) Change step 12.9 (test instrument removal) to reflect changes in test equipment as described above.

6) Changes to Section 13 (Enclosures)

   a) Add Enclosures 13.9 and 13.10 for dLOG Setup.

   b) Change Enclosure 13.1 to reflect changes in test equipment.

   c) Delete Note on Enclosure 13.1 allowing blanks to be N/A'd if test equipment not used (test equipment must be used with dLOG).

   d) Change Enclosure 13.4 to require only one reading instead of three because dLOG can be used to obtain one accurate average value for a five minute period.

The following is a list of changes associated with the deletion of the requirement to take data at "max flow", thus making 600 gpm the highest flow data point.

1) Change the range on the discharge pressure gauge from 2000 psig to 1800 psig.

2) Delete the need for a 0-800 inwd D/P gauge.

3) Delete Notes prior to steps 8.1.1 and 12.7, and delete former step 8.1.2, concerning installation of the 0-800 inwd gauge.

4) Add Note prior to step 12.7 to avoid over-ranging of instrumentation (since the 0-800 inwd D/P gauge will no longer be used). Also, change the acceptance range for the 600 gpm flow point from (575-625 gpm) to (575-610 gpm) to avoid instrument over-ranging.

The option of taking vibration data was also deleted in this procedure retype because such data is obtained during each IWP test. The following changes were made to reflect this:

1) Delete need for vibration probe in Sections 5 and 10 (Test Equipment and Data Required, respectively).

2) Change the last sentence of the Test Method Section (Section 9) to indicate that no vibration data is required, and to explain the option of performing manual calculations, or using dLOG.

3) Delete Vibration data Note in Enclosure 13.4.

The remainder of changes included in the retype are listed below. Many of them reflect an attempt to standardize the head curve procedures (as much as possible) between units. Others are changes required to ensure that successful completion of the procedure can be achieved with minimal discrepancies.

1) The following References were added to Section 2:

   a) OP/1/A/6250/02, Auxiliary Feedwater System
   b) OP/1/A/6450/18, CA Pump Pit CO2 System

2) Add Limits and Precautions 6.5 and 6.6 to make sure pump miniflow limits are observed, and pump bearing temperatures are monitored by the Control Room Operator.

3) Change wording of CO2 and Firewatch steps to be consistent with the IWP procedures. The wording used (or to be used) for these steps in the IWP procedures is consistent with Operations' recommendations, and was developed through the Cross-Disciplinary Review process. Also remove the Independent Verification (I.V.) requirement for Operations to take the CO2 system out of service and return it back to service. The I.V. is not required because all actions taken are covered under OP/1/A/6450/18, CA Pump Pit CO2 System.

4) Add phrase "Have Operations..." to the beginning of each step dealing with the valve alignment.

5) Add Note concerning CA system operability to follow the step which has Operations align valves to their "Required Initial Test Position". This change was originated through a step in the two-year procedure review process.

6) Add step following step 12.7.1 to have the flow throttled to approximately 400 gpm before the pump is shut off. This is to allow Operations to readily complete their Head and Valve Verification procedure at the completion of this test.

7) Add Note following step 12.16 to ensure that the appropriate documents (Flow Balance Procedure, and Unit 1 Data Book) are updated if the % pump calculation result differs from the previous such result.

8) Add step to ensure that the Control Room Senior Reactor Operator and the Control Room Operator are informed of the completion of this test.

9) Change process gauge numbers in Enclosure 13.4 to the correct numbers.

10) Valve 1CA88, CA Pump 1B Discharge to S/G Isol, was added to the valve alignment to be required in the open position. This will further ensure proper system configuration during the performance of this test.

The purpose of this procedure is to verify that CA Pump 1B is operating within acceptable limits, and to provide a pump strength value for the CA Flow Balance Procedure (PT/1/A/4250/03E).

The changes associated with the incorporation of the dLOG Computer Application for data acquisition do not alter the Test Method used. They do, however, change the method that data is acquired by utilizing a benchmarked computer data acquisition system to take data and perform calculations. The instruments used still meet the required accuracies stated in the CA System Test Acceptance Criteria Data Sheet CNTC-1592-CA-V002-03 (Rev. 1).

Other changes included in this retype delete the requirements to take vibration data, and head data at flows greater than 600 gpm. Although pump vibration is a critical parameter which should be monitored carefully, the deletion of the requirement to obtain vibration data via this procedure will not affect our pump performance monitoring program. This data is obtained regularly per the quarterly IWP procedure, PT/1/A/4250/03B. Limiting the range of flow rates at which Total Dynamic Head data is acquired to less than or equal to 600 gpm will not degrade the effectiveness of our pump performance monitoring. This range amply covers the required flow rates of CA Pump 1B in the event of an accident.

No Unreviewed Safety Questions are created by these changes.

PT/2/A/4250/13A Retype, Changes 0 to 4 Incorporated

The previously unapproved changes included in this retype are intended to ensure as much similarity as possible between the Unit 1 and Unit 2 procedures. The intent of each of the major changes included in this retype can be described by one of the following:

- Incorporation of the use of the benchmarked dLOG computer program for data acquisition into this procedure.

- Deletion of the requirement to take Head data at "max flow". (600 gpm is now the highest flow rate at which a data point is required).

- Deletion of requirement to trend pump motor bearing temperatures during this test. This activity is not within the scope of the main objective of this procedure, and is performed annually per the IWP procedure for Auxiliary Feedwater (CA) Pump 2A (PT/2/4250/03A). A Limit and Precaution which informs the Control Room Operator to monitor pump motor bearing temperatures was added to this procedure in place of the trending requirement.

- Addition/Deletion of various steps to ensure proper Administrative Controls during the performance of this procedure.

The purpose of this procedure is to verify that CA Pump 2A is operating within acceptable limits, and to provide a pump strength value for the CA Flow Balance Procedure (PT/2/A/4250/03E).

The changes associated with the incorporation of the dLOG Computer Application for data acquisition do not alter the Test Method used. They do, however, change the method that data is acquired by utilizing a benchmarked computer data acquisition system to take data and perform calculations. The instruments used still meet the required accuracies stated in the CA System Test Acceptance Criteria Data Sheet CNTC-2592-CA-V001-03 (Rev. 0).

Other changes included in this retype delete the requirements to trend pump motor bearing temperature data, and to take head data at flows greater than 600 gpm. Although pump motor bearing temperature is a critical parameter which should be monitored carefully, the deletion of the requirement to obtain vibration data via this procedure will not affect our pump performance monitoring program. This data is trended on an annual basis per the IWP procedure, PT/2/A/4250/03A. Limiting the range of flow rates at which Total Dynamic Head data is acquired to less than or equal to 600 gpm will not degrade the effectiveness of our pump performance monitoring. This range amply covers the required flow rates of CA Pump 2A in the event of an accident.

All other changes included in this retype are part of an attempt to maintain similarity between the Unit 1 and Unit 2 procedures.

No Unreviewed Safety Questions are created as a result of these changes.

TN/1/A/1239/00/01A, Initial Issue

The purpose of Nuclear Station Modification (NSM) CN-11239 is to install a pressure relief path to prevent overpressurization of Penetration M-322. This NSM will install a check valve and piping around Safety Injection (NI) valve 1NI095A to provide this pressure relief.

This procedure will control the installation of the piping and valves and assure Post Modification Testing is completed to maintain system integrity. Part of the NI system will be isolated to perform this modification while the unit is in modes 5, 6, or No Mode. Steps in TN/1/A/1239/00/01A insure that containment integrity will be maintained during Core Alterations, per Tech. Spec. 3/4.9.4, by isolating the NI System within the boundaries of valves 1NI120B, 1NI096B, and 1NI360. The Operations group will control the isolations for implementation and testing. An unreviewed safety question is not created by the implementation of this procedure.

MP/0/A/7450/45, Initial Issue

This procedure has been written to provide guidance in performing corrective maintenance on the Control Area Chilled Water (YC) Chillers.

Maintenance performed under this procedure has been reviewed against approved vendor manuals, design documents, and station procedures to ensure that corrective maintenance controlled by this procedure will return this chiller to as-built/as-designed condition. These actions will ensure the chiller's compliance with FSAR accident analysis. Since the chiller will be returned to

as-designed conditions, the possibility, consequences or probability of a malfunction will be reduced. Therefore, no Unreviewed Safety Question exists.

MP/0/A/7150/88, Initial Issue

This procedure has been written to provide a method to take ice condenser baskets apart and put them back together again should it become necessary during outage work.

This procedure has been written in a manner as to maintain the as-designed, as-built condition of the ice baskets as shown on drawing CNM-1201.17-30 and as described in the Westinghouse Ice Basket Installation Procedure CNM-1201.17-343.

This procedure has been written based on approved vendor manuals, design documents, and station procedures to ensure that the actions controlled will return the ice baskets to their as-designed/as-built condition. These actions will ensure the ice basket's compliance with FSAR accident analysis. Since the ice baskets will be returned to as-designed conditions, the possibility, probability or consequences of a malfunction will be reduced. Therefore, no Unreviewed Safety Question exists.

PT/1/A/4450/13E, Initial Issue

During the investigation of PIR 2-C90-156 it was found that the solenoids on the three carbon dioxide pilot control valves were installed backwards. To keep this problem from occurring in the future, it was recommended that a nitrogen pressure test of the pilot control valves be performed to verify that they do not leak through. It was also recommended that each pilot control valve be actuated in order to verify that pilot pressure is received at the selector valve.

In order to perform this additional testing, this new procedure (PT/1/A/4450/13E) was written. This new procedure will be performed every three years. It will verify that the pilot control valves hold pressure and that the selector valves operate, and also will verify that the piping from the discharge header to the nozzles in the pits is unobstructed. Nitrogen will be used to test the pilot control valves. Instrument air (VI) will be blown out the discharge nozzles to verify piping is unobstructed.

Although the Auxiliary Feedwater (CA) CO2 system is not directly nuclear safety related, it does protect equipment that is. Chapter 16.9-3 of the FSAR (Selected Licensee Commitment Manual) requires that the CA CO2 system be operable whenever equipment protected by it is required to be operable. (The CA system is required to be operable in modes 1, 2, and 3.) During this test, the CA CO2 system will be inoperable, but a continuous firewatch will be established as required by the Selected Licensee Commitment Manual action for CO2 system out of service. There is also a fire hose station located in the CA pump room, so that at no time will any of the equipment protected by the CA CO2 system be left unprotected.

The CA CO2 system does not directly affect the safe shutdown of the plant. No safety related equipment is left unprotected. No conditions of this test violate the design basis assumptions as described in the FSAR. The margin of safety as defined in the Tech. Spec. bases is not reduced. No unreviewed safety question is created.

TT/0/A/9100/57, Initial Issue

PIR 0-C90-279 was written because testing of the fire hose stations was inadequate to ensure design flow and pressure was being delivered. The fire hose system is designed to deliver a flow of 75-100 gpm at a minimum pressure of 65 psig. It was recommended that several of the fire hose stations in the Auxiliary Building be flow tested to ensure that no degradation had occurred that would affect the fire protection system.

In order to perform this testing, this temporary test (TT/0/A/9100/57) was written. This test will involve full flow testing several hose stations throughout the Auxiliary Building. The water will be discharged into Groundwater Drainage System (WZ) Sump C. For the hose stations on the 543' elevation, pressure and flowrate data will be taken. For the other elevations, a restrictive orifice will be inserted between the end of the hose and the discharge to the drain, and the pressure back at the hose station valve will be monitored. The orifice will restrict the flow to a range of 75-100 gpm through the pressure range of 60-100 psig.

All the fire hose stations that are to be tested are required by Chapter 16.9-4 of the FSAR (Selected Licensee Commitment Manual) to be operable whenever equipment that is protected by them is required to be operable. During this test, the hose station being tested will be inoperable due to its normal fire hose being disconnected and test hose being connected between the hose station and the discharge drain. There is a Caution statement at the beginning of each test that says that the fire hose should be disconnected from the hose station for less than an hour, or some means of protection for the area left unprotected will be provided. Since each hose station will be inoperable for less than an hour, and since there will be someone at the hose station valve to shut it and reconnect the fire hose in case of a fire, none of the equipment protected by the hose stations will be left unprotected.

During the performance of this test, it will be necessary to keep open certain fire doors and tornado pressure doors. In order to do this, the Compensatory Action Guidelines for Plant Access Doors must be followed. This document gives the requirements for closing these doors and the time in which the action must be performed.

Since this test does not affect any other parts of the Fire Protection System and there are sufficient compensatory actions in place to return those parts of the system the test does affect to their normal conditions, the probability of and consequences of an accident, whether or not previously evaluated in the FSAR is not increased. Since none of the equipment protected by the hose stations is left unprotected, the probability of and consequences of a malfunction of a piece of equipment important to safety, whether or not previously evaluated in the FSAR, is not increased. Since the conditions of this test do not violate the design basis assumptions as described in the

FSAR, the margin of safety as defined in the basis is not reduced. No unreviewed safety question is created.

PT/2/A/4150/13E, Initial Issue

This procedure will be used to determine the value for tempering flow to be input into the Operator Aid Computer (OAC) Thermal Outputs Program when Auxiliary Feedwater nozzle tempering flow is established.

Method:

The test will be performed in 3 parts:

1) Data will be recorded from the OAC for 2 hours to establish a baseline.

2) Tempering flow will be established and another 2 hours of data will be recorded. With the 2 sets of data the tempering flow can be calculated.

   Since during the second run Secondary Power will be invalid, Main Turbine First Stage Pressure will be used to detect changes in power level between the two runs. This has been determined to be the most repeatable and most stable indicator of Power Level using many 2 hour data runs on the OAC.

   The Main Feedwater (CF) Venturi Flow in the first run is corrected to the power level of the second run. The difference between the corrected CF Venturi Flow from the first run and the CF Venturi Flow from the second run is the tempering flow for that Steam Generator.

   These calculated tempering flows are then input as constants into the OAC Thermal Output Program.

3) Data will be recorded for another 2 hours. Then the Secondary Power Level will be compared to the first run Power Level after correcting for Main Turbine First Stage Pressure changes. The difference between the two will be verified to be within 0.1% of each other.

Safety Evaluation

Having Tempering flow on or off has no effect on the operation of Unit except for the calculation of Reactor Power. During the second 2 hour run, Secondary Power will be in error by the amount of the tempering flow. During the entire test Reactor Power will be limited to 99% or less. The Control Room Operator will be informed of the Error in Secondary Thermal Power during the second run. He will be told to monitor Primary Thermal Power and not to let Power to increase above the level measured during the first run. This will ensure that during the test that 100% Reactor Power will not be exceeded.

The latest Secondary Power Uncertainty Analysis is Design Calculation CNC-1552.08-00-0019 dated 7/31/87. In this analysis, tempering flow is accounted for even though Unit 2 normally has tempering flow isolated. The assumed Uncertainty for tempering flow is 45 gpm per S/G. This equates to a 90 GPM or [45 * SQRT(4)] uncertainty for total unit tempering flow. This

determination of tempering flow in this procedure is well within this uncertainty. Therefore, establishing tempering flow and calculating the tempering flow rate per this procedure will not increase the uncertainty of the OAC Thermal Power Calculations.

No unreviewed safety questions are created.

PT/2/A/4250/13B, Initial Issue

The intent of this procedure is to ensure that Auxiliary Feedwater (CA) Pump 2B is operating within acceptable limits and to provide a pump strength value for the CA System Flow Balance Procedure (PT/2/A/4250/03E).

This procedure is used to ensure that in the event of an accident, the pump strength of CA Pump 2B is acceptable to fulfill its design function. This procedure actually decreases the probability of safety equipment malfunction by allowing any notable decrease in the strength of CA Pump 2B to be recognized through testing. This test method used involves normal operating/testing processes (manual start, alignment to the Upper Surge Tank, etc.).

Although this procedure may be performed in modes 1, 2, or 3 when the CA System is required, the margin of safety as defined by Tech. Specs. is not reduced because the remaining two CA Pumps are assumed to be available and able to fulfill the required function of the CA System during the performance of this test.

PT/1/A/4350/15A Retype, Changes 0 to 20 Incorporated

The following changes have been incorporated into this procedure reissue.

In Section 12.3, the steps that specify jumpers to be placed to simulate an emergency start have been changed to specify a momentary placement of the jumpers. Section 12.4 has been broken into several subsections. A step has been added to Section 12.4.4 to ensure that all isolation valves to the lube oil pressure switches are open when testing is finished. A caution statement has also been added to 12.4.4 stating that if the TRIP LOW PRESSURE LUBE OIL alarm is received, the operator is to depress the ENABLE NON-EMER TRIP reset pushbutton. This will allow the Diesel Generator (D/G) to trip while the low-low pressure lube oil trip is being tested.

The Diesel Generator 1A Periodic Test is performed in accordance with Tech. Specs. 4.8.1.1.2.g.3, 4.8.1.1.2.g.2, 4.8.1.1.2.g.5, 4.8.1.1.2.g.7, 4.8.1.1.2.g.6c, 4.8.1.1.2.g.11, and 4.8.1.1.2.g.13. The following criteria are verified:

1) The capability of the D/G to reject a load of 825 kw while maintaining voltage at 4160 ± 420 volts, and frequency at 60 ± 1.2 Hz.

2) The capability of the D/G to reject a load of $\geq$ 5600 kw but $\leq$ 5750 kw without tripping and without allowing D/G speed to exceed 500 rpm.

3) All automatic D/G trips, except engine overspeed, low-low lube oil pressure, generator differential, and the 2 out of 3 voltage controlled overcurrent relay scheme are bypassed upon loss of voltage on the emergency bus concurrent with a Safety Injection Actuation signal.

4) The Fuel Oil transfer valve transfers fuel from each fuel storage tank to the day tank of the D/G.

5) The barring device engaged and MAINTENANCE mode lockout features prevent the D/G from starting.

6) The D/G will start on an auto-start signal.

7) The capability of the D/G to carry a load of > 5600 kw but ≤ 5750 kw for ≥ 24 hours.

During performance of this test, the D/G will remain operable except during Sections 12.3 and 12.4. In Section 12.3 the D/G will be placed in MAINTENANCE mode and the barring device will be engaged, both of which prevent the D/G from starting. In Section 12.4, the Lo-Lo Lube Oil Pressure trip is blocked. The overall test method remains the same as in the previous issue of this procedure.

Only D/G 1A is made inoperable by this test; D/G 1B is still available to supply the B-Train 4160V essential bus. The D/G 1A will be run in a normal alignment, both paralleled to the grid and in an idling condition, by approved operating procedures. The accident analysis assumes only one train is available. If D/G 1A were to fail, D/G 1B will supply its essential bus. No unreviewed safety question is created.

PT/2/A/4350/15A Retype, Changes 0 to 9 Incorporated

The following changes have been incorporated into this procedure reissue.

A step has been added to Section 12.4.4 to ensure that all isolation valves to the lube oil pressure switches are open when testing is finished. A caution statement has also been added to 12.4.4 stating that if the TRIP LOW PRESSURE LUBE OIL alarm is received, the operator is to depress the ENABLE NON-EMER TRIP reset pushbutton. This will allow the D/G to trip while the low-low pressure lube oil trip is being tested. Steps 12.3.20 and 12.3.25 were added to ensure Non-Emergency trips are reinstated. Editorial changes were also made which do not affect the test method.

The Diesel Generator 2A Periodic Test is performed in accordance with Tech. Specs. 4.8.1.1.2.g.3, 4.8.1.1.2.g.2, 4.8.1.1.2.g.5, 4.8.1.1.2.g.7, 4.8.1.1.2.g.6c, 4.8.1.1.2.g.11, and 4.8.1.1.2.g.13. The following criteria are verified:

1) The capability of the D/G to reject a load of 825 kw while maintaining voltage at 4160 ± 420 volts, and frequency at 60 ± 1.2 Hz.

2) The capability of the D/G to reject a load of ≥ 5600 kw but ≤ 5750 kw without tripping and without allowing D/G speed to exceed 500 rpm.

3) All automatic D/G trips, except engine overspeed, low-low lube oil pressure, generator differential, and the 2 out of 3 voltage controlled overcurrent relay scheme are bypassed upon loss of voltage on the emergency bus concurrent with a Safety Injection Actuation signal.

4) The Fuel Oil transfer valve transfers fuel from each fuel storage tank to the day tank of the D/G.

5) The barring device engaged and MAINTENANCE mode lockout features prevent the D/G from starting.

6) The D/G will start on an auto-start signal.

7) The capability of the D/G to carry a load of > 5600 kw but ≤ 5750 kw for ≥ 24 hours.

During performance of this test, the D/G will remain operable except during Sections 12.3 and 12.4. In Section 12.3, the D/G will be placed in MAINTENANCE mode and the barring device will be engaged, both of which prevent the D/G from starting. In Section 12.4, the Lo-Lo Lube Oil Pressure trip is blocked. The overall test method remains the same as in the previous issue of this procedure. Section 12.4 has been broken into several subsections.

Only D/G 2A is made inoperable by this test; D/G 2B is still available to supply the B-Train 4160V essential bus. The D/G 2A will be run in a normal alignment, both paralleled to the grid and in an idling condition, by approved operating procedures. The accident analysis assumes only one train is available. If D/G 2A were to fail, D/G 2B will supply its essential bus. No unreviewed safety question is created.

PT/2/A/4350/15B Retype, Changes 0 to 9 Incorporated

The following changes have been incorporated into this procedure reissue.

A step has been added to Section 12.4.4 to ensure that all isolation valves to the lube oil pressure switches are open when testing is finished. A caution statement has also been added to 12.4.4 stating that if the TRIP LOW PRESSURE LUBE OIL alarm is received, the operator is to depress the ENABLE NON-EMER TRIP reset pushbutton. This will allow the Diesel Generator (D/G) to trip while the low-low pressure lube oil trip is being tested. Steps 12.3.20 and 12.3.25 were added to ensure Non-Emergency trips are reinstated. Editorial changes were also made which do not affect the test method.

The Diesel Generator 2B Periodic Test is performed in accordance with Tech. Specs. 4.8.1.1.2.g.3, 4.8.1.1.2.g.2, 4.8.1.1.2.g.5, 4.8.1.1.2.g.7, 4.8.1.1.2.g.6c, 4.8.1.1.2.g.11, and 4.8.1.1.2.g.13. The following criteria are verified:

1) The capability of the D/G to reject a load of 825 kw while maintaining voltage at 4160 ± 420 volts, and frequency at 60 ± 1.2 Hz.

2) The capability of the D/G to reject a load of ≥ 600 kw but ≤ 5750 kw without tripping and without allowing D/G speed to exceed 500 rpm.

3) All automatic D/G trips, except engine overspeed, low-low lube oil pressure, generator differential, and the 2 out of 3 voltage controlled overcurrent relay scheme are bypassed upon loss of voltage on the emergency bus concurrent with a Safety Injection Actuation signal.

4) The Fuel Oil transfer valve transfers fuel from each fuel storage tank to the day tank of the D/G.

5) The barring device engaged and MAINTENANCE mode lockout features prevent the D/G from starting.

6) The D/G will start on an auto-start signal.

7) The capability of the D/G to carry a load of > 5600 kw but ≤ 5750 kw for ≥ 24 hours.

During performance of this test, the D/G will remain operable, except during Sections 12.3 and 12.4. In Section 12.3, the D/G will be placed in MAINTENANCE mode and the barring device will be engaged, both of which prevent the D/G from starting. In Section 12.4, the Lo-Lo Lube Oil Pressure trip is blocked. The overall test method remains the same as in the previous issue of this procedure.

Only D/G 2B is made inoperable by this test; D/G 2A is still available to supply the A-Train 4160V essential bus. The D/G 2A will be run in a normal alignment, both paralleled to the grid and in an idling condition, by approved operating procedures. The accident analysis assumes only one train is available. If D/G 2B were to fail, D/G 2A will supply its essential bus. No unreviewed safety question is created.

PT/1/A/4350/15B Retype, Changes 0 to 20 Incorporated

The following changes have been incorporated into this procedure reissue.

In Section 12.3, the steps that specify jumpers to be placed to simulate an emergency start have been changed to specify a momentary placement of the jumpers. Section 12.4 has been broken into several subsections. A step has been added to Section 12.4.4 to ensure that all isolation valves to the lube oil pressure switches are open when testing is finished. A caution statement has also been added to 12.4.4 stating that if the TRIP LOW PRESSURE LUBE OIL alarm is received, the operator is to depress the ENABLE NON-EMER TRIP reset pushbutton. This will allow the Diesel Generator (D/G) to trip while the low-low pressure lube oil trip is being tested.

The Diesel Generator 1B Periodic Test is performed in accordance with Tech. Specs. 4.8.1.1.2.g.3, 4.8.1.1.2.g.2, 4.8.1.1.2.g.5, 4.8.1.1.2.g.7, 4.8.1.1.2.g.6c, 4.8.1.1.2.g.11, and 4.8.1.1.2.g.13. The following criteria are verified:

1) The capability of the D/G to reject a load of 825 kw while maintaining voltage at 4160 ± 420 volts, and frequency at 60 ± 1.2 Hz.

2) The capability of the D/G to reject a load of ≥ 5600 kw but ≤ 5750 kw without tripping and without allowing D/G speed to exceed 500 rpm.

3) All automatic D/G trips, except engine overspeed, low-low lube oil pressure, generator differential, and the 2 out of 3 voltage controlled overcurrent relay scheme are bypassed upon loss of voltage on the emergency bus concurrent with a Safety Injection Actuation signal.

4) The Fuel Oil transfer valve transfers fuel from each fuel storage tank to the day tank of the D/G.

5) The barring device engaged and MAINTENANCE mode lockout features prevent the D/G from starting.

6) The D/G will start on an auto-start signal.

7) The capability of the D/G to carry a load of > 5600 kw but ≤ 5750 kw for ≥ 24 hours.

During performance of this test, the D/G will remain operable except during Sections 12.3 and 12.4. In Section 12.3 the D/G will be placed in MAINTENANCE mode and the barring device will be engaged, both of which prevent the D/G from starting. In Section 12.4, the Lo-Lo Lube Oil Pressure trip is blocked. The overall test method remains the same as in the previous issue of this procedure.

Only D/G 1B is made inoperable by this test; D/G 1A is still available to supply the A-Train 4160V essential bus. The D/G will be run in a normal alignment, both paralleled to the grid and in an idling condition, by approved operating procedures. The accident analysis assumes only one train is available. If D/G 1B were to fail, D/G 1A will supply its essential bus. No unreviewed safety question is created.

MP/0/A/7150/05 Retype, Changes 0 to 6 Incorporated

The following changes were made to the procedure during the re-write:

- The applicable FSAR section was added to the References Section 2.0.
- References to Health Physics in Section 4.2 were changed to Radiation Protection.
- Designated Personnel in the Hold part of Steps 6.1 and 6.2 were changed to Maint. Rep.
- Maintenance Engineer in Step 6.3 was changed to MES Rep.
- Step 6.4 was changed from a step to a note.
- Step 8.2 dealing with the inspection of lifting devices has been removed. This function is now performed in the Tool Room.

- The Accep ce Requirements of Section 9.0 have been changed to reflect the new higher minimum acceptable ice basket weight. Technical Specification 3/4.6.5.1 has been changed to allow the weighing surveillance interval to increase from 9 to 18 months. In increasing the weighing interval, the minimum allowable ice basket weight for Tech. Specs. has increased. The minimum acceptable weights shown in Section 9.0 reflect the new total ice basket weights for baskets with and without cable cruciforms.

- MES Mechanical Engineer in Steps 11.3.1.3 was changed to MES Rep.

- Step 11.3.4.3 was changed to specify clevis pin and to include the keeper.

- Step 11.3.4.11 was changed from "Install pin" to "Install clevis pin and keeper".

- Verifier in Steps 6.1 and 6.2 of Enclosure 13.1 Section A was changed to Maint. Rep.

- MES Mechanical Engineer in Steps 11.3.1.3 and 11.3.2 of Enclosure 13.1 Section B was changed to MES Rep.

- The Data Sheet Page 2 of 5 of Enclosure 13.1 has been neatened up. The acceptance requirements have been changed to reflect Section 9.0.

- The Data Sheet Page 3 of 5 of Enclosure 13.1 has been changed to eliminate the unnecessary signoff blanks. Step 13.3.3.12 of this sheet has been changed to include the clevis pin keepers in the final inspection.

- Step 11.4.2 of Enclosure 13.1 page 4 of 5 has been changed to show the new acceptance requirements of Section 9.0.

- Step 8.2 of Enclosure 13.1 has been deleted.

Technical Specification Section 4.6.5.1.b.2 addresses the surveillance requirements for weighing ice baskets. FSAR Section 6.7.4, Ice Baskets, addresses the design, fabrication, and installation of ice baskets.

This procedure is based on approved vendor manuals, design documents, and station procedures and has been compared with Tech. Specs. to ensure that the actions controlled by this procedure will comply with established surveillance requirements. This procedure will ensure that accurate ice basket weights are taken on randomly selected baskets which will be used to determine ice bed operability. Since the procedure will accurately determine basket weights on randomly selected baskets, the possibility, consequences, or probability of a malfunction will be reduced. Therefore, no unreviewed safety question exists.

TN/1/B/0632/00/02A, Initial Issue

Nuclear Station Modification (NSM) CN-10632, Rev. 0 will modify the ice condenser piping associated with the floor cooling units on the Ice Condenser

Refrigeration (NF) System. This modification will be implemented in parts. TN/1/B/0632/00/02A provides guidance for modifying piping in the lower elevations of the NF system.

The primary concern for implementation of this modification is melting the ice in the ice condenser. With this consideration in mind, this modification will be implemented in parts to minimize the time that the NF system will be out of service. TN/1/B/0632/00/02A involves work activities associated with the lower elevation tie-ins and modificatio of the NF system. This work will be performed during a Unit 1 outage whil th Maintenance Ice Condenser crew is on their scheduled off time. This will enable the NF floor cooling units to be removed from service while minimizing the possibility of ice melting.

Technical Specification 3/4.6.5 discusses the ice condenser. This Technical Specification is only required for Modes 1, 2, 3, and 4 and requires that the temperature be maintained below 27 degrees. Implementation of this modification will not cause the Ice Bed Temperature Monitoring System to be inoperable; therefore, Operations will still be capable of monitoring the temperature of the ice condenser. In addition, this modification will not affect the Ice Bed Doors or Door Position Monitoring System or any other system associated with the NF System.

Section 6.7 of the FSAR discusses the Ice Condenser design. NSM CN-10632 will reroute the ice condenser floor cooling supply line from the AHU return line directly to the lower portion of the ice condenser compartments. This will enable the NF System to provide cooler glycol to the floor cooling units and will decrease the temperature differential between the lower ice condenser and the ice bed.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/B/0632/00/03A, Initial Issue

Nuclear Station Modification (NSM) CN-10632, Rev. 0 will modify the ice condenser piping associated with the floor cooling units on the Ice Condenser Refrigeration (NF) System. This modification will be implemented in parts. TN/1/B/0632/00/03A provides guidance for modifying piping in the upper elevations of the NF system.

The primary concern for implementation of this modification is melting the ice in the ice condenser. With this consideration in mind, this modification will be implemented in parts to minimize the time that the NF system will be out of service. TN/1/B/0632/00/03A involves work activities associated with the upper elevation tie-ins and modifications of the NF system. This work will be performed during a Unit 1 outage while the Maintenance Ice Condenser crew is on their scheduled off time. This will enable the NF system to be removed from service while minimizing the possibility of ice melting.

Technical Specification 3/4.6.5 discusses the ice condenser. This Technical Specification is only required for Modes 1, 2, 3 and 4 and requires that the temperature be maintained below 27 degrees. Implementation of this modification will not cause the Ice Bed Temperature Monitoring System to be

inoperable; therefore, Operations will still be capble of monitoring the temperature of the ice condenser. In addition, this modification will not affect the Ice Bed Doors or Door Position Monitoring System or any other system associated with the NF System.

Section 6.7 of the FSAR discusses the Ice Condenser design. NSM CN-10632 will reroute the ice condenser floor cooling supply line from the AHU return line directly to the lower portion of the ice condenser compartments. This will enable the NF System to provide cooler glycol to the floor cooling units and will decrease the temperature differential between the lower ice condenser and the ice bed.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/3175/CE/01A, Initial Issue

Exempt Change CE-3175 authorizes the deletion of the snubbers for Component Cooling (KC) system supports 1-A-KC-4108 and 1-A-KC-4177. This procedure provides guidance for the removal of the snubbers from supports referenced above.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the KC system piping during implementation of this procedure. Design Engineering has provided instructions for the implementation of this modification. In accordance with these instructions, implementation of this work unit does not make the KC system inoperable. All support deletions may be completed with the affected KC system piping in service, and KC system operability will not be affected.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/2929/CE/01A, Initial Issue

Exempt Change CE-2929 will modify the control circuit for Residual Heat Removal (ND) Valve 1ND002A (ND Pump 1A Suction from Reactor Coolant Loop B) so the interlock with valves 1FW027A (ND Pump 1A Suction from Refueling Water Storage Tank) and 1NS043A (ND Pump 1A Discharge to Containment Spray Header) will be dependent on valve position only. The modification will also modify the control circuit for valve 1ND036B (ND Pump 1B Suction from Reactor Coolant Loop C) so the interlock with valves 1FW055B (ND Pump 1B Suction from Refueling Water Storage Tank) and 1NS038B (ND Pump 1B Discharge to Containment Spray Header) will be dependent on valve position only. Presently, removal of power from either 1FW027A or 1NS043A will prevent 1ND002A from opening and removal of power from either 1FW055B or 1NS038B will prevent 1ND036B from opening.

This procedure will control the work on the A-train valves associated with CE-2929. Construction Maintenance Department (CMD) will perform all work remote to the valves and Instrument and Electrical Group (IAE) will perform

all work on the valves. No new equipment or components will be added by this modification.

This procedure will be implemented with Unit 1 in Modes 4, 5 or 6 with ND Train B operable or with Unit 1 in No Mode.

IAE will rewire and setup the rotors on valves 1FW027A and 1NS043A and perform a functional verification on the valves in accordance with IP/0/A/3820/04. Performance will stroke time valves 1FW027A and 1NS043A to verify operability. An unreviewed safety question does not exist.

TN/1/A/2929/CE/02, Initial Issue

Exempt Change CE-2929 will mod fy the control circuit for Residual Heat Removal (ND) Valve 1ND002A (ND Pump 1A Suction from Reactor Coolant Loop B) so the interlock with valves 1FW027A (ND Pump 1A Suction from Refueling Water Storage Tank) and 1NS043A (ND Pump 1A Discharge to Containment Spray Header) will be dependent on valve position only. The modification will also modify the control circuit for valve 1ND036B (ND Pump 1B Suction from Reactor Coolant Loop C) so the interlock with valves 1FW055B (ND Pump 1B Suction from Refueling Water Storage Tank) and 1NS038B (ND Pump 1B Discharge to Containment Spray Header) will be dependent on valve position only. Presently, removal of power from either 1FW027A or 1NS043A will prevent 1ND002A from opening and removal of power from either 1FW055B or 1NS038B will prevent 1ND036B from opening.

This procedure will control the work on the B-train valves associated with CE-2929. Construction Maintenance Department (CMD) will perform all work remote to the valves and Instrument and Electrical Group (IAE) will perform all work on the valves. No new equipment or components will be added by this modification.

This procedure will be implemented with Unit 1 in Modes 4, 5 or 6 with ND Train A operable or with Unit 1 in No Mode.

IAE will rewire and setup the rotors on valves 1FW055B and 1NS038B and perform a functional verification on the valves in accordance with IP/0/A/3820/04. Performance will stroke time valves 1FW055B and 1NS038B to verify operability. An unreviewed safety question does not exist.

TN/5/A/0414/00/01A, Initial Issue

The purpose of this procedure is to add piping and valves to the Nuclear Service Water (RN) strainer backwash lines for use during the addition of clamtrol into the RN pump pits. The addition of this modification will prevent the release of treated water to Lake Wylie during clamtrol treatment. The RN strainers backflush automatically on a time cycle unless overridden by a pre-set high pressure drop. Internal water pressure is the motive force for dislodging strained particles, as a backflush drive motor turns a backwash arm past the various strainer assemblies. The discharge is released to atmospheric pressure and dumps into a trash basket outside the RN Pumphouse. Entrained trash is collected, and the water is returned to the Standby Nuclear

Service Water Pond, which overflows to Lake Wylie. The addition of piping and isolation valves by this modification will not affect the strainer backflush process. However, it will assure that during clamtrol addition the backflush will be routed through the new piping and strainer basket for entrained trash collection. The water will be returned to the RN Pump pit, instead of being routed to the trash pits on the outside of the RN Pumphouse. This will assure that the clamtrol treated water will remain in the pump pits and not be released to the Standby Nuclear Service Water Pond. At such time clamtrol is not needed, the backflush operations can continue as they are presently operating.

Scaffolding must be erected in the RN pump pits for the implementation of this modification. Design Engineering has evaluated the operability concerns and has provided Engineering Instructions to assure operability during scaffold erection and modification implementation. The RN Pumphouse is a Class 1 seismically designed structure that contains two separate pits, from which two independent and redundant channels of RN pumps take suction. This procedure requires that the scaffolding be erected according to Design Engineering instructions. Even in the occurrence of an unusual event, based on the Engineering Instructions depiciting the type of materials required for the erection of the scaffolding, as well flow characteristics in the pump pits, and lattice screens to protect the RN pumps from solid objects, the pump pits would remain operable.

The RN System A-Train wil be out of service during the tie-ins for this modification. Tech. Spec. 3/4.7.4 requires that the RN System A-Train be returned to service within 72 hours. The RN System provides essential auxiliary support functions to Engineered Safety Features of the station. The RN System is designed to supply cooling water to various heat loads in both the safety and non-safety portions of each unit, and provisions are made to ensure continuous flow of cooling water to those systems and components necessary for plant safety during normal operation and under accident conditions. Sufficient redundancy of piping and components is provided to ensure that cooling is maintained to essential loads at all times.

Post Modification Testing (PMT) will be conducted in accordance with PMT Letter and established procedures. There are no abnormal isolations for Post Modification Testing.

This procedure will control the implementation of NSM CN50414 which will enhance the system by safely allowing the addition of clamtrol, and does not degrade any design parameters, and cannot initiate any FSAR accidents. There are no reviewed safety questions associated with this procedure.

TN/5/A/0414/00/02A, Initial Issue

The purpose of this procedure is to add piping and valves to the Nuclear Service Water (RN) strainer backwash lines for use during the addition of clamtrol into the RN pump pits. The addition of this modification will prevent the release of treated water to Lake Wylie during clamtrol treatment. The RN strainers backflush automatically on a time cycle unless overridden by a pre-set high pressure drop. Internal water pressure is the motive force for dislodging strained particles, as a backflush drive motor turns a backwash arm

past the various strainer assemblies. The discharge is released to atmospheric pressure and dumps into a trash basket outside the RN Pumphouse. Entrained trash is collected, and the water is returned to the Standby Nuclear Service Water Pond, which overflows to Lake Wylie. The addition of piping and isolation valves by this modification will not affect the strainer backflush process. However, it will assure that during clamtrol addition the backflush will be routed through the new piping and strainer basket for entrained trash collection. The water will be returned to the RN Pump pit, instead of being routed to the trash pits on the outside of the RN Pumphouse. This will assure that the clamtrol treated water will remain in the pump pits and not be released to the Standby Nuclear Service Water Pond. At such time clamtrol is not needed, the backflush operations can continue as they are presently operating.

Scaffolding must be erected in the RN pump pits for the implementation of this modification. Design Engineering has evaluated the operability concerns and has provided Engineering Instructions to assure operability during scaffold erection and modification implementation. The RN Pumphouse is a Class 1 seismically designed structure that contains two separate pits, from which two independent and redundant channels of RN pumps take suction. This procedure requires that the scaffolding be erected according to Design Engineering instructions. Even in the occurrence of an unusual event, based on the Engineering Instructions depicting the type of materials required for the erection of the scaffolding, as well flow characteristics in the pump pits, and lattice screens to protect the RN pumps from solid objects, the pump pits would remain operable.

The RN System B-Train will be out of service during the tie-ins for this modification. Tech. Spec. 3/4.7.4 requires that the RN System A-Train be returned to service within 72 hours. The RN System provides essential auxiliary support functions to Engineered Safety Features of the station. The RN System is designed to supply cooling water to various heat loads in both the safety and non-safety portions of each unit, and provisions are made to ensure continuous flow of cooling water to those systems and components necessary for plant safety during normal operation and under accident conditions. Sufficient redundancy of piping and components is provided to ensure that cooling is maintained to essential loads at all times.

Post Modification Testing (PMT) will be conducted in accordance with PMT Letter and established procedures. There are no abnormal isolations for Post Modification Testing.

This procedure will control the implementation of NSM CN50414 which will enhance the system by safely allowing the addition of clamtrol, and does not degrade any design parameters, and cannot initiate any FSAR accidents. There are no reviewed safety questions associated with this procedure.

TN/1/A/1216/00/01A, Initial Issue

This procedure will replace Steam Generator 1D Blowdown Isolation (BB) Valve 1BB010B with a new gate valve, Item #06H-210. This is being performed due to numerous maintenance problems and marginally sized operators.

The flow path from Steam Generator 1D will be out of service during the replacement of valve 1BB010B. The containment isolation valves upstream of 1BB010B will be utilized to satisfy Tech. Spec. requirements for control of penetrations which have direct access to outside atmosphere for containment integrity/closure during core alterations, fuel movement, and Reactor Coolant System mid-loop conditions. Breaker F07A in 1EMXL will be opened and Red Tagged to ensure valve 1BB010B is de-energized for electrical work. Also, sliding links will be opened to isolate the monitor light for valve 1BB010B. These isolations do not present a concern for the safe operation of the BB system, because the BB system is not required to be operable in modes 5, 6, or no mode. Red tags will be removed, and breaker F07A in 1EMXL and sliding links in 1EATC2 will be closed to allow for valve set up, verification of remote position indication, Motor Operated Valve (MOV) and stroke time testing. The installation and set up for new valve 1BB010B will be complete prior to entry into mode 4. A post modification review will be performed prior to mode 4 to satisfy containment integrity requirements and allow further testing that must be performed in modes 1 through 4.

Testing of the new valve will consist of performing MOV Testing and verifying all remote position, and status light indications. Stroke time tests will be performed prior to mode 3, and again in mode 3. A differential pressure and flow verification tests will be performed on valve 1BB010B. Hydrostatic and appropriate Non-Destructive Examination (NDE) tests will also be performed. All of this post modification testing will be performed using approved station procedures.

No new failure modes are being introduced. The safety related aspects of the BB system will be maintained during implementation of this work unit.

Based on the above discussion, it is determined that an Unreviewed Safety Question does not exist.

TN/1/A/1216/00/02A, Initial Issue

This procedure will replace Steam Generator 1A Blowdown Isolation (BB) Valve 1BB057B with a new gate valve, Item #06H-210. This is being performed due to numerous maintenance problems and marginally sized operators.

The flow path from Steam Generator 1A will be out of service during the replacement of valve 1BB057B. The containment isolation valves upstream of 1BB057B will be utilized to satisfy Tech. Spec. requirements for control of penetrations which have direct access to outside atmosphere for containment integrity/closure during core alterations, fuel movement, and Reactor Coolant System mid-loop conditions. Breaker F07B in 1EMXL will be opened and Red Tagged to ensure valve 1BB057B is de-energized for electrical work. Also, sliding links will be opened to isolate the monitor light for valve 1BB057B. These isolations do not present a concern for the safe operation of the BB system, because the BB system is not required to be operable in modes 5, 6, or no mode. Red tags will be removed, and breaker F07B in 1EMXL and sliding links in 1EATC2 will be closed to allow for valve set up, verification of remote position indication, Motor Operated Valve (MOV) and stroke time testing. The installation and set up for new valve 1BB057B will be complete prior to entry into mode 4. A post modification review will be performed

prior to mode 4 to satisfy containment integrity requirements and allow further testing that must be performed in modes 1 through 4.

Testing of the new valve will consist of performing MOV Testing and verifying all remote position and status light indications, as well as testing of the newly installed anti-hammer circuit. Stroke time tests will be performed prior to mode 3, and again in mode 3. A differential pressure and flow verification tests will be performed on valve 1BB057B. Hydrostatic and appropriate Non-Destructive Examination (NDE) tests will also be performed. All of this post modification testing will be performed using approved station procedures.

No new failure modes are being introduced. The safety related aspects of the BB system will be maintained during implementation of this work unit.

Based on the above discussion, it is determined that an Unreviewed Safety Question does not exist.

TN/1/A/1216/00/03A, Initial Issue

This procedure will replace Steam Generator 1B Blowdown Isolation (BB) Valve 1BB021B with a new gate valve, Item #06H-210. This is being performed due to numerous maintenance problems and marginally sized operators.

The flow path from Steam Generator 1B will be out of service during the replacement of valve 1BB021B. The containment isolation valves upstream of 1BB021B will be utilized to satisfy Tech. Spec. requirements for control of penetrations which have direct access to outside atmosphere for containment integrity/closure during core alterations, fuel movement, and Reactor Coolant System mid-loop conditions. Breaker R02D in 1EMXB will be opened and Red Tagged to ensure valve 1BB021B is de-energized for electrical work. Also, sliding links will be opened to isolate the monitor light for valve 1BB021B. These isolations do not present a concern for the safe operation of the BB system, because the BB system is not required to be operable in modes 5, 6, or no mode. Red tags will be removed, and breaker R02D in 1EMXB and sliding links in 1EATC5 will be closed to allow for valve set up, verification of remote position indication, Motor Operated Valve (MOV) and stroke time testing. The installation and set up for new valve 1BB021B will be complete prior to entry into mode 4. A post modification review will be performed prior to mode 4 to satisfy containment integrity requirements and allow further testing that must be performed in modes 1 through 4.

Testing of the new valve will consist of performing MOV Testing and verifying all remote position and status light indications. Stroke time tests will be performed prior to mode 3, and again in mode 3. A differential pressure and flow verification tests will be performed on valve 1BB021B. Hydrostatic and appropriate Non-Destructive Examination (NDE) tests will also be performed. All of this post modification testing will be performed using approved station procedures.

No new failure modes are being introduced. The safety related aspects of the BB system will be maintained during implementation of this work unit.

Based on the above discussion, it is determined that an Unreviewed Safety Question does not exist.

TN/1/A/1216/00/04A, Initial Issue

This procedure will replace Steam Generator 1C Blowdown Isolation (BB) Valve 1BB061B with a new gate valve, Item #06H-210. This is being performed due to numerous maintenance problems and marginally sized operators.

The flow path from Steam Generator 1C will be out of service during the replacement of valve 1BB061B. The containment isolation valves upstream of 1BB061B will be utilized to satisfy Tech. Spec. requirements for control of penetrations which have direct access to outside atmosphere for containment integrity/closure during core alterations fuel movement, and Reactor Coolant System mid-loop conditions. Breaker F02B in 1EMXB will be opened and Red Tagged to ensure valve 1BB061B is de-energized for electrical work. Also, sliding links will be opened to isolate the monitor light for valve 1BB061B. These isolations do not present a concern for the safe operation of the BB system, because the BB system is not required to be operable in modes 5, 6, or no mode. Red tags will be removed, and breaker F02B in 1EMXB and sliding links in 1EATC5 will be closed to allow for valve set up, verification of remote position indication. Motor Operated Valve (MOV) and stroke time testing. The installation and set up for new valve 1BB061B will be complete prior to entry into mode 4. A post modification review will be performed prior to mode 4 to satisfy containment integrity requirements and allow further testing that must be performed in modes 1 through 4.

Testing of the new valve will consist of performing MOV Testing and verifying all remote position and status light indications, as well as testing of the newly installed anti-hammer circuit. Stroke time tests will be performed prior to mode 3, and again in mode 3. A differential pressure and flow verification tests will be performed on valve 1BB061B. Hydrostatic and appropriate Non-Destructive Examination (NDE) tests will also be performed. All of this post modification testing will be performed using approved station procedures.

No new failure modes are being introduced. The safety related aspects of the BB system will be maintained during implementation of this work unit.

Based on the above discussion, it is determined that an Unreviewed Safety Question does not exist.

PT/1/A/4250/14 Retype, Changes 0 to 8 Incorporated

This procedure retype consolidates the four sections of the existing procedure into two sections in Enclosure Format. To do so, an "Autostart" section, along with a "Manual start" section, were constructed, and the option was given to align the system per either the "Miniflow" or "Flow other than Miniflow" alignment Enclosure. Neither the test method nor the actual steps in this procedure are significantly affected. Also, the steps associated with all data parameters are now optional, so that a particular section may be performed without having to obtain unnecessary data. Descriptive notes were

placed on many of these steps so that the Test Coordinator can make a quick and accurate assessment of whether or not to N/A a step, and the test could be performed more effectively.

A third main section is being added to this procedure as a result of this retype (T&T Valve Start and Stop, Enclosure 13.3). This section allows only a Visicorder trace to be obtained, and has a separate Acceptance Criteria (11.3) to verify that Auxiliary Feedwater Pump Turbine (CAPT) #1 can be started and stopped using the Trip and Throttle (T&T) valve.

This procedure verifies proper operation of the CAPT #1 governor and associated equipment. The restructuring of this procedure does not significantly affect the test method nor the actual steps within the procedure. The T&T Valve Start/Stop section was added by this retype to provide a method of ensuring that CAPT #1 will operate properly when controlled by the T&T Valve only (as required during a safe shutdown facility (SSF) event). The addition of the T&T Valve Start/Stop section will allow problems with the T&T Valve to be detected through testing rather than during an actual event. This test will only be performed after permission has been granted by the Operations shift personnel, who control the operation of the CA Pumps, to ensure that the two Motor-Driven CA pumps are available in the event that an accident occurs during the performance of this test.

Therefore, no Unreviewed Safety Questions exist as a result of this procedure retype.

PT/2/A/4250/14 Retype, Changes 0 to 2 Incorporated

This procedure retype consolidates the four sections of the existing procedure into two sections in Enclosure Format. To do so, an "Autostart" section, along with a "Manual start" section, were constructed, and the option was given to align the system per either the "Miniflow" or "Flow other than Miniflow" alignment Enclosure. Neither the test method nor the actual steps in this procedure are significantly affected. Also, the steps associated with all data parameters are now optional, so that a particular section may be performed without having to obtain unnecessary data. Descriptive notes were placed on many of these steps so that the Test Coordinator can make a quick and accurate assessment of whether or not to N/A a step, and the test could be performed more effectively.

A third main section is being added to this procedure as a result of this retype (T&T Valve Start and Stop, Enclosure 13.3). This section allows only a Visicorder trace to be obtained, and has a separate Acceptance Criteria (11.3) to verify that Auxiliary Feedwater Pump Turbine (CAPT) #2 can be started and stopped using the Trip and Throttle (T&T) valve.

This procedure verifies proper operation of the CAPT #2 governor and associated equipment. The restructuring of this procedure does not significantly affect the test method nor the actual steps within the procedure. The T&T Valve Start/Stop section was added by this retype to provide a method of ensuring that CAPT #2 will operate properly when controlled by the T&T Valve only (as required during a Safe Shutdown Facility (SSF) event). The addition of the T&T Valve Start/Stop section will allow

problems with the T&T Valve to be detected through testing, rather than during an actual event. This test will only be performed after permission has been granted by the Operations shift personnel, who control the operation of the CA Pumps, to ensure that the two Motor-Driven CA pumps are available in the event that an accident occurs during the performance of this test.

Therefore, no Unreviewed Safety Questions exist as a result of this procedure retype.

OP/0/B/6500/09 Revision 4, Changes 0 to 32 Incorporated

Catawba Nuclear Station (CNS) Technical Specification 3/4.11.3 requires that all radioactive wastes be solidified or dewatered in accordance with the Process Control Program (PCP) to meet shipping and transportation requirements during transit, and disposal site requirements when received at the disposal site. This Technical Specification implements the requirements of 10CFR50.36a and General Design Criterion 60 of Appendix A to 10CFR50. The Duke Power Company (DPC) PCP was developed to assure that solidification and dewatering activities are performed in compliance with 10CFR20, 50, 61, 71 and 49CFR. The PCP states:

> "Vendor procedures shall be incorporated as attachments to station procedures. Vendor format may be retained as a DPC enclosure if desired, or the procedure may be rewritten into DPC format."

At Catawba, the vendor format is retained, and new enclosures are issued as major procedure changes whenever a vendor procedure is revised and reissued. These new enclosures cover the dewatering of powdered resin mixtures and the handling of high integrity container overpacks, and have been reviewed and approved by the following areas:

- Chem-Nuclear Systems, Inc. Safety Review Board

- Duke Power Company, Nuclear Production Department, Nuclear Technical Services (General Office)

The new enclosures do not sigificantly affect structures, systems, or components that are addressed in the FSAR. Spent resins are transferred to the liners using approved Duke Power Company procedures. The Chem-Nuclear procedures are used only for the dewatering of the material after it is placed in the liner, and the handling of the liner.

The Catawba systems that interface with vendor equipment are:

- Station Air (VS)

- Makeup Demineralized Water Sytsem (YM)

- Solid Radwaste System (WS)

This revision: 1) incorporates four changes (Changes #29 through Change #32), 2) replaces three revised pages (#4, #5, and #9) of Enclosure 4.10, <u>Process Control Program for the CNSI Demineralization Systems DM-OP-025</u>. (These three

pages were reissued to correct "psig" to "psid" on two pages and correct "OSOTOPIC" to "ISOTOPIC" on the other page) and 3) restructures the format on steps 3.1.6 and 3.1.7.

No unreviewed safety question is created by these changes.

PT/1/A/4400/03D Retype, Changes 0 to 1 Incorporated

This test balances flow to Component Cooling (KC) supplied components under the Refueling and Engineered Safety Features (ESF) modes as described in the FSAR. When performing the Refueling balance in modes 5 and 6, Residual Heat Removal (ND) System operability is required and will be maintained by ensuring that KC flow stays above 5000 gpm. In no mode, neither train is required; therefore, the Tech. Specs. are not affected. For ESF mode, the test will be on a single train at a time; therefore, the associated components will be declared inoperable per Tech. Specs. An unreviewed safety question does not exist.

OP/1/A/6350/01 Retype #3, Changes 11 to 15 Incorporated

The changes made to this procedure affect the system in a significant manner by removing the equipment from service, but they do not affect the evaluation made in the FSAR. FSAR sections 8.1.4 and 8.3.1 were reviewed. The major change to the procedure is the addition of enclosures to remove and return the following from/to service:

ETA/ETB
FTA/FTB
6900 Volt Switchgear
Train A (B) of Main Power

The enclosures are written so that any required load is transferred to another power source before its normal power source is removed. These enclosures are being added as a result of the study done due to Catawba Incident Investigation Report C88-006-2. Unit 2 was changed to an earlier date. The Unit 1 procedure has been improved due to things learned from the Unit 2 procedure.

By using these enclosures, an undesired train blackout can be prevented, since the sequencer is removed from service before de-energizing ETA or ETB. The enclosures will remove one train of equipment at a time. This will only be done during No Mode when no Tech. Specs. will be affected.

In the past, all the actions performed in these enclosures were handled by a worklist item and tagouts. The procedure will provide the guidelines for removing and restoring this equipment, which will require a review by more than one person to change the order of the procedure or any other change required.

It is judged that the changes made in this retype do not create an unreviewed safety question.

MP/0/A/7150/91A, Initial Issue

This procedure is for removal and replacement of the Reactor Coolant (NC) Pump Motor Flywheel to perform flywheel inservice inspections.

This procedure has been reviewed against approved vendor manuals, design documents, and station procedures to ensure that activities controlled by this procedure will return to Flywheel and the NC Pumps to the as-designed condition. Regulatory Guide 1.14 was reviewed for compliance. The flywheel is removed using this procedure to perform an examination of all exposed surfaces and a complete ultrasonic volumetric examination. This procedure meets requirements of Reg. Guide 1.14 to insure integrity of the Flywheel through inservice inspection. Since this procedure maintains the Flywheel and the NC Pump in their as-design condition, the possibility, consequences, or probability of a malfunction will be reduced. Therefore, no unreviewed safety question exists.

MP/0/A/7200/05 Retype, Changes 0 to 2 Incorporated

The procedure title has been changed to give a bett.. description of what the procedure includes. Section 11.0 of this procedure has been revised as follows:

- Section 11.29 has steps 21, 22 and 23 added to provide more detail for correctly installing the trip rod after maintenance has been performed.

- Section 11.30 was revised to include steps for electrical or mechanical overspeed trip testing. It was also revised to insure that maintenance monitors the bearing oil temperatures of the turbine during testing.

- Enclosure 13.4 was replaced with the current revision from the vendor.

- Enclosure 13.8 was revised to show a more complete tool list.

Tech. Spec. 3/4.7.1 is affected by this procedure. Operations has the responsibility and the procedures for compliance with these Tech. Specs. Maintenance will be performed on this turbine when Tech. Specs. allow, per Operations' procedures. This rewrite will clarify and assure that maintenance activities will return the turbines to as-designed conditions.

The changes made by this rewrite have been reviewed against approved vendor manuals, design documents, and station procedures to ensure that the corrective maintenance controlled by this procedure will return the turbine to as-built/as-designed condition. These actions will ensure the turbine's compliance with FSAR accident analysis. Since the turbine will be returned to as-designed conditions, the possibility, consequences, or probability of a malfunction will be reduced. Therefore, no unreviewed safety question exists.

MP/0/B/7600/123, Initial Issue

This procedure performs corrective maintenance on valves 1 (2) CF-9 and -16, Atwood and Morrill 24" Check Valves with side air cylinders. This procedure is only applicable to non-safety related equipment.

This procedure has been reviewed against approved vendor manuals, design documents, and station procedures to ensure that the corrective maintenance controlled by this procedure will return the valve to as-built/as-designed condition. These actions will ensure valve compliance with FSAR accident analysis. Therefore, no unreviewed safety question exists.

TN/1/A/1107/01/03A, Initial Issue

Nuclear Station Modification (NSM) CN-11107, Rev. 0 provides an alternate recirculation loop for the Unit 1 Boric Acid Tank (BAT) in the chemical control, purification, and makeup subsystem of the Chemical and Volume Control (NV) system. A recirculation pump with isolation valves, drain valves, and instrumentation will be added for BAT recirculation. The pump will employ suction and discharge from existing BAT connections. This new recirculation loop will provide a more thorough and uniform mixing of the BAT contents than is presently provided by the existing Boric Acid Transfer pumps.

In order to perform this procedure, the Unit 1 BAT must be drained to facilitate system tie-in's. The BAT supplies boric acid solution to support unit operation. The tank is designed to store sufficient boric acid solution for a cold shutdown from full power operation. The BAT will be drained while Unit 1 is in NO MODE; therefore, unit shutdown requirements will not be a concern. After completion of the BAT tie-in's, the BAT pressure boundary will be restored and the BAT can be returned to service to provide make-up capabilities to support unit operation as required by the FSAR.

This modification also requires system tie-in's to the Boron Recycle (NB) and Liquid Waste (WL) systems. The NB tie-in will be performed using a wet-tap procedure to eliminate system down time. No NB system isolations are required, and the wet-tap will not affect NB system operation. Therefore, no additional operational concerns are imposed. The WL tie-in will be installed in a Class 'E' section of piping, which supports various station equipment drains and leak-off flows. The tie-in is located at an elevation above the drain header; therefore, installation will not impact system draining capabilities.

This procedure installs the BAT recirculation pump, recirculation piping, along with manual isolation valves and check valves, and makes the required tie-in's to the BAT, NB and WL piping. After installation, portions of the piping will be hydrostatically tested and the remainder visually inspected at system temperature and pressure to verify piping integrity.

No new failure modes or operating characteristics are introduced by this procedure. All systems affected by this procedure will be able to perform their intended functions. Based on the considerations above, no unreviewed safety questions are judged to be involved or created by this procedure.

PT/2/A/4250/03D Retype, Changes 0 to 18 Incorporated

This procedure is used to verify the response times of valves associated with the Nuclear Service Water (RN) assured water source to suction of the Auxiliary Feedwater (CA) pumps. The changes involved in this retype have all been previously approved.

Since the CA System is not required in Modes 4, 5, or 6, no accident scenarios are impacted. Since all changes included in this retype have previously been approved, the margin of safety as defined in the design basis is unaffected. Also, neither the probability, nor the consequence of safety equipment malfunction is increased. No unreviewed safety question is created.

PT/1/A/4250/03D Retype, Changes 0 to 33 Incorporated

The following is a summary of previously unapproved changes included in this retype. The intent of this retype is to ensure that the test can be performed with minimal discrepancies, and to maintain similarity between the Unit 1 and Unit 2 procedures.

1) The following to Section V (Test Equipment) was added to ensure that the Test Coordinator is properly prepared to run the test:

   "_ Key to Unit 1 Auxiliary Shutdown Panel
   _ link slider
   _ wrench
   _ Four jumpers (with labelled tags)"

2) "1EATC5" was replaced with "the backside of 1EATC5" in Steps 12.2.5.6, 12.2.5.7, 12.2.6.8, 12.2.6.9, 12.2.7.6 12.2.7.7, and 12.2.7.21. This change clarified existing procedure steps.

3) A note was added prior to all steps requiring the Auxiliary Feedwater (CA) System Auto-Start Defeat to be "DEFEATED". The Note will enable each of these steps to be N/A, initialed, and dated if Solid State Protection System (SSPS) (for the appropriate Train) is in "TEST" Mode. This change is necessary because the CA System Auto-Start Defeat will not go into "DEFEAT" mode if the corresponding train of SSPS is in "TEST".

4) In the sections where a Motor-Driven CA Pump Auto-Start signal is simulated, steps were added to ensure that the motor breaker for the appropriate pump is opened prior to the completion to the section. Although the motor breakers are "Racked to Test" (for the pump receiving the Auto-Start signal), these steps will serve as an additional precautionary measure after the pump motor breaker closes on the Auto-Start signal. Note: the alignments ensure that both motor breakers are returned to their "As Found" positions following completion of the corresponding section.

5) Due to the change discussed above (Item 4), the second notes following Steps 12.2.7.19 and 12.1.7.19 were deleted because they no longer apply.

6) A note was added to precede Step 12.2.8 so that Steps 12.2.8 and 12.2.9 may be performed out of sequence. Due to the locations of the links

involved in these steps, this change will allow the procedure to be performed in a more simple, logical manner.

7) Step 12.5 (and the note accompanying it) were added to ensure that the test is logged out of the Unit 1 Test Logbook, and the key to the Unit 1 Auxiliary Shutdown Panel is returned to Operations upon completion of the test.

This procedure is used to verify the response times of valves associated with the Nuclear Service Water (RN) assured water source to suction of the CA pumps.

Items 1, 2, and 6 are administrative controls to ensure that the procedure can be followed accurately and with minimal problems. Item 3, concerning the CA System Auto-Start Defeat, ensures that the affected procedure steps can be completed under varying system configurations (i.e., SSPS in TEST or in NORMAL).

Items 4 and 5 ensure that proper system configuration is restored after the motor driven pump motor breakers close on auto-start signals.

Since the CA System is not required in Modes 4, 5, or 6, no accident scenarios are impacted. Also, the margin of safety as defined in the design basis is unaffected. Neither the consequences, nor the probability of safety equipment malfunction are increased. No reviewed safety question is created.

TN/1/A/1146/00/01A, Initial Issue

Nuclear Station Modification (NSM) CN-11146 Rev. 0 will make the following changes: 1) Delete the Nuclear Service Water (RN) piping that supplies flush water for the Ventilation Unit Condensate Drain Tank (VUCDT) Radiation Monitors (EMFs). 2) Provide new piping from the Component Cooling (KC) system to supply the Auxiliary Shutdown Panel Supply Units (ASPSUs) with an additional RN supply tie in that will be used during an Auxiliary Shutdown Panel (ASP) event. 3) Delete all the old RN piping that used to supply the ASPSUs. 4) Provide a flow path to flush out the RN piping that supplies the Auxiliary Feedwater (CA) pumps. This flow path will tie into the Condenser Circulating Water system, bypassing the CA system.

NSM CN-11146 Rev. 0 has been broken up into five Work Units for implementation purposes. Work Unit 01 will perform the following: 1) Cut and cap the old RN piping which used to supply the A Train ASPSU. The RN piping will be cut and capped at the 6" supply and 20" return headers. A Post Modification work request will be used to remove the majority of this old RN piping and associated support restraints. 2) A new tie in to the 6" RN A Train piping will be made including a new drain and isolation valve. 3) New 4" piping will also be tied into the 6" RN A Train piping. This will be part of the new "Clam Flush" piping which will tie into the RC system under Work Unit 03. The piping to be installed under Work Unit 01 will include the A Train isolation valve and all the piping up to the support restraints specified by Design Engineering.

The isolations required to perform the work asociated with this procedure fall under the scope of routine tagouts performed by the operational control group. The piping being deleted/installed per this procedure will have pipe caps or isolation valves that will provide for system isolation. To further ensure system isolation and operability, the isolation valves will be tagged closed and all the piping supports required for system and seismic integrity will be installed per Design Engineering instructions. Design has analyzed this particular installation configuration and approved its implementation. The freeze seals required for hydrostatic testing will be controlled by an approved station procedure. All of the isolations and testing associated with this procedure have been analyzed or will be controlled by existing approved procedures. No new failure modes are being introduced.

The safety related aspects of the RN, CA, and KC systems will be maintained during implementation of this Work Unit.

Based on the above discussion, there are no unreviewed safety questions associated with this procedure.

TN/1/A/1146/00/02A, Initial Issue

Nuclear Station Modification (NSM) CN-11146 Rev. 0 will make the following changes: 1) Delete the Nuclear Service Water (RN) piping that supplies flush water for the Ventilation Unit Condensate Drain Tank (VUCDT) Radiation Monitors (EMFs). 2) Provide new piping from the Component Cooling (KC) system to supply the Auxiliary Shutdown Panel Supply Units (ASPSUs) with an additional RN supply tie in that will be used during an Auxiliary Shutdown Panel (ASP) event. 3) Delete all the old RN piping that used to supply the ASPSUs. 4) Provide a flow path to flush out the RN piping that supplies the Auxiliary Feedwater (CA) pumps. This flow path will tie into the Condenser Circulating Water (RC) system, bypassing the CA system.

NSM CN-11146 Rev. 0 has been broken up into five Work Units for implementation purposes. Work Unit 02 will perform the following: 1) Delete the remaining RN piping going to the VUCDT EMFs and cap the pipe. 2) Tie into the KC A Train Essential Supply and Return headers. 3) Install piping and other components from the KC system tie-ins up to ASPSU-1A. 4) Tie into the RN ASPSU-1A supply line installed under Work Unit 01. 5) Install ASPSU-1A outlet piping up to and including valve 1KCD30 (the RC system isolation valve.).

The isolations required to perform the work asociated with this procedure fall under the scope of routine tagouts performed by the operational control group. The piping being deleted/installed per this procedure will have pipe caps or isolation valves that will provide for system isolation. To further ensure system isolation and operability, the isolation valves will be tagged closed and all the piping supports required for system and seismic integrity will be talled per Design Engineering instructions. Design has analyzed this particular installation configuration and approved its implementation. The freeze seals required for hydrostatic testing will be controlled by an approved station procedure. All of the isolations and testing associated with this procedure have been analyzed or will be controlled by existing approved procedures.

The safety related aspects of the RN, CA, and KC systems will be maintained during implementation of this Work Unit.

Based on the above discussion, there are no unreviewed safety questions associated with this procedure.

TN/1/A/1146/00/03A, Initial Issue

Nuclear Station Modification (NSM) CN-11146 Rev. 0 will make the following changes: 1) Delete the Nuclear Service Water (RN) piping that supplies flush water for the Ventilation Unit Condensate Drain Tank (VUCDT) Radiation Monitors (EMFs). 2) Provide new piping from the Component Cooling (KC) system to supply the Auxiliary Shutdown Panel Supply Units (ASPSUs) with an additional RN supply tie in that will be used during an Auxiliary Shutdown Panel (ASP) event. 3) Delete all the old RN piping that used to supply the ASPSUs. 4) Provide a flow path to flush out the RN piping that supplies the Auxiliary Feedwater (CA) pumps. This flow path will tie into the Condenser Circulating Water (RC) system, bypassing the CA system.

NSM CN-11146 Rev. 0 has been broken up into five Work Units for implementation purposes. Work Unit 03 will perform the following: 1) Make all the tie-ins to the piping installed under Work Units 01, 02, 04, and 05. 2) Make the tie-in to the RC system. 3) Install a new 2" CA drain line. 4) Perform all the necessary Post Modification Testing required for CN-11146 Rev. 0.

The isolations required to perform the work asociated with this procedure fall under the scope of routine tagouts performed by the operational control group. The piping being deleted/installed per this procedure will have pipe caps or isolation valves that will provide for system isolation. Since Work Unit 03 installs all the remaining piping, supports valves, and other components associated with CN-11146 Rev. 0, the seismic and pressure boundary integrity of the RN, CA, RC, and KC systems will be ensured. The Post Modification Testing to be performed under this procedure will verify flows through the piping installed under all five Work Units. Adequate operation of the ASPSUs will also be verified using RN and KC supplied water. The valve alignments and operation of related equipment will be controlled using the procedure for Work Unit 03 and any Operational Control Group procedures required for system alignments and equipment operation. The Post Modification Testing does not put the RN, KC, CA, or RC systems into an unusual alignment. The testing will use flow paths and alignments that will be used during normal and ASP event system operation. The KC supply to the ASPSUs is considered to be minimal relative to the total KC essential header supply flow. Even with a maximum of 10 GPM going to the ASPSU, the effect on the flow being supplied to the Residual Heat Removal (ND) pumps motor coolers is considered to be negligible. Therefore, even if the ND pumps are required to be operable during the period of time that the post modification testing is being performed, the pumps will still have sufficient flow to operate adequately and provide their safety related function as required by Tech. Specs.

A KC flow balance will not be necessary to ensure KC or ND operability due to the Post Modification Testing being performed in this procedure prior to Mode 4. A KC flow balance will be performed prior to entering Mode 4 up to ensure KC system operability required for Mode 4 plant operation.

All of the isolations and testing associated with this procedure have been analyzed or will be controlled by existing approved procedures. The safety related aspects of the RN, CA, and KC systems will be maintained during implementation of this Work Unit.

Based on the above discussion, there are no unreviewed safety questions associated with this procedure.

TN/1/A/1146/00/04A, Initial Issue

Nuclear Station Modification (NSM) CN-11146 Rev. 0 will make the following changes: 1) Delete the Nuclear Service Water (RN) piping that supplies flush water for the Ventilation Unit Condensate Drain Tank (VUCDT) Radiation Monitors (EMFs). 2) Provide new piping from the Component Cooling (KC) system to supply the Auxiliary Shutdown Panel Supply Units (ASPSUs) with an additional RN supply tie in that will be used during an Auxiliary Shutdown Panel (ASP) event. 3) Delete all the old RN piping that used to supply the ASPSUs. 4) Provide a flow path to flush out the RN piping that supplies the Auxiliary Feedwater (CA) pumps. This flow path will tie into the Condenser Circulating Water (RC) system, bypassing the CA system.

NSM CN-11146 Rev. 0 has been broken up into five Work Units for implementation purposes. Work Unit 04 will perform the following: 1) Cut and cap the old RN piping which used to supply the B Train ASPSU. The RN piping will be cut and capped at the 20" return header. A Post Modification work request will be used to remove the majority of this old RN piping and associated support restraints. 2) A new tie into the 6" RN B Train piping will be made including a new drain and isolation valve. 3) New 4" piping will also be tied into the 6" RN B Train piping at the location of the old 2" RN supply to the ASPSU. This will be part of the new "Clam Flush" piping which will tie into the RC system under Work Unit 03. The piping to be installed under Work Unit 04 will include the B Train isolation valve and all the piping up to the support restraints specified by Design Engineering.

The isolations required to perform the work asociated with this procedure fall under the scope of routine tagouts performed by the operational control group. The piping being deleted/installed per this procedure will have pipe caps or isolation valves that will provide for system isolation. To further ensure system isolation and operability, the isolation valves will be tagged closed and all the piping supports required for system and seismic integrity will be installed per Design Engineering instructions. Design has analyzed this particular installation configuration and approved its implementation. The freeze seals required for hydrostatic testing will be controlled by an approved station procedure. All of the isolations and testing associated with this procedure have been analyzed or will be controlled by existing approved procedures. No new failure modes are being introduced.

The safety related aspects of the RN, CA, and KC systems will be maintained during implementation of this Work Unit.

Based on the above discussion, there are no unreviewed safety questions associated with this procedure.

TN/1/A/1146/00/05A, Initial Issue

Nuclear Station Modification (NSM) CN-11146 Rev. 0 will make the following changes: 1) Delete the Nuclear Service Water (RN) piping that supplies flush water for the Ventilation Unit Condensate Drain Tank (VUCDT) Radiation Monitors (EMFs). 2) Provide new piping from the Component Cooling (KC) system to supply the Auxiliary Shutdown Panel Supply Units (ASPSUs) with an additional RN supply tie in that will be used during an Auxiliary Shutdown Panel (ASP) event. 3) Delete all the old RN piping that used to supply the ASPSUs. 4) Provide a flow path to flush out the RN piping that supplies the Auxiliary Feedwater (CA) pumps. This flow path will tie into the Condenser Circulating Water (RC) system, bypassing the CA system.

NSM CN-11146 Rev. 0 has been broken up into five Work Units for implementation purposes. Work Unit 05 will perform the following: 1) Tie into the KC A Train Essential Supply and Return headers. 2) Install piping and other components from the KC system tie-ins up to ASPSU-1B. 3) Tie into the RN ASPSU-1B supply line installed under Work Unit 04. 5) Install ASPSU-1B outlet piping up to and including valve 1KCD39 (the RC system isolation valve.)

The isolations required to perform the work asociated with this procedure fall under the scope of routine tagouts performed by the operational control group. The piping being deleted/installed per this procedure will have pipe caps or isolation valves that will provide for system isolation. To further ensure system isolation and operability, the isolation valves will be tagged closed and all the piping supports required for system and seismic integrity will be installed per Design Engineering instructions. Design has analyzed this particular installation configuration and approved its implementation. The freeze seals required for hydrostatic testing will be controlled by an approved station procedure. All of the isolations and testing associated with this procedure have been analyzed or will be controlled by existing approved procedures. No new failure modes are being introduced.

The safety related aspects of the RN, CA, and KC systems will be maintained during implementation of this Work Unit.

Based on the above discussion, there are no unreviewed safety questions associated with this procedure.

TN/1/A/0058/00/06A Retype #1, Change 01 Incorporated

This procedure is for the installation of new 2" Stainless Steel (SS) Nuclear Service Water (RN) piping from supply valve 1RNC92 through Diesel Generator Starting Air (VG) Aftercooler 1B2 to return valve 1RNC54. This new piping will allow greater unobstructed flow through VG Aftercooler 1B2. The existing 1" Carbon Steel piping will be removed from the system, with valve 1RN964 being deleted and valve 1RN961 remaining as a system drain.

Currently the VG Aftercoolers are supplied by the RN System through 1" Carbon Steel piping, which has become partially restricted due to corrosion. The replacement of this piping with 2" Stainless Steel piping will provide the required flow to the aftercoolers and will not be as susceptible to corrosion.

Implementing this procedure will require isolating and draining of the portion of the RN System that supplies VG Aftercooler 1B2. The Operations Group will coordinate the isolations necessary to implement this procedure. This procedure will be implemented during Modes 5 and 6 with 'A' Diesel operable, or during No Mode of the unit refueling outage. VG Aftercooler 1B2 will be out of service during this modification. The piping and equipment affected by this procedure can be out of servie during the modes and conditions specified above.

Post Modification Testing will include a hydro-static test and a visual inspection for leakage. These tests will assure the pressure boundaries and integrity of the system are maintained. Also, performance will test the new piping to assure the flow and pressure drop capabilities of the aftercooler have improved. The implementation of this procedure does not degrade any design parameters, and cannot initiate any FSAR accidents. The implementation of this procedure will not cause any new failure modes. There is no increase in the probability or consequences of a malfunction of equipment important to safety previously evaluated in the FSAR. Based on this discussion, the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR is not created.

Since no safety parameters, setpoints, or design limits have been adversely affected, no margin is safety as defined in the bases to any Technical Specification is reduced. Based on the above, there are no unreviewed safety questions associated with this procedure.

TN/1' '0058/00/03A Retype #1, Change 01 Incorporated

This procedure is for the installation of new 2" Stainless Steel (SS) Nuclear Service Water (RN) piping from supply valve 1RNC90 through Diesel Generator Starting Air (VG) Aftercooler 1A2 to return valve 1RNC52. This new piping will allow greater unobstructed flow through VG Aftercooler 1A2. The existing 1" Carbon Steel piping will be removed from the system, with valve 1RN953 being deleted and valve 1RN950 remaining as a system drain.

Currently the VG Aftercoolers are supplied by the RN System through 1" Carbon Steel piping, which has become partially restricted due to corrosion. The replacement of this piping with 2" Stainless Steel piping will provide the required flow to the aftercoolers and will not be as susceptible to corrosion.

Implementing this procedure will require isolating and draining of the portion of the RN System that supplies VG Aftercooler 1A2. The Operations Group will coordinate the isolations necessary to implement this procedure. This procedure will be implemented during Modes 5 and 6 with 'B' Diesel operable, or during No Mode of the unit refueling outage. VG Aftercooler 1A2 will be out of service during this modification. The piping and equipment affected by this procedure can be out of service during the modes and conditions specified above.

Post Modification Testing will include a hydro-static test and a visual inspection for leakage. These tests will assure the pressure boundaries and integrity of the system are maintained. Also, Performance will test the new piping to assure the flow and pressure drop capabilities of the aftercooler

have improved. The implementation of this procedure does not degrade any design parameters, and cannot initiate any FSAR accidents. The implementation of this procedure will not cause any new failure modes. There is no increase in the probability or consequences of a malfunction of equipment important to safety previously evaluated in the FSAR. Based on this discussion, the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR is not created.

Since no safety parameters, setpoints, or design limits have been adversely affected, no margin is safety as defined in the bases to any Technical Specification is reduced. Based on the above, there are no unreviewed safety questions associated with this procedure.

TN/1/A/2678/CE/01A, Initial Issue

This procedure provides guidance for the implementation of Exempt Change CE-2678, Work Unit 01A. This modification replaces manual loader 1VYML0190 with a Moore Products 3528 Single Loop Digital Controller. This manual loader provides manual control of valve 1VY19, which is adjusted to regulate air flow from containment into the annulus when the Containment Hydrogen Purge System is being used. The new controller will be programmed to function as a manual loader, thereby providing the same function as the existing manual loader.

All work in this procedure will be performed with Unit 1 in Modes 5, 6, or No Mode. The Containment Hydrogen Purge System is not required during these modes. The manual loader is located on Main Control Board 1MC5. This panel will not be adversely affected by the work performed in this procedure and no other equipment will be affected. Electrical isolations for this procedure involve opening/closing sliding links and will only affect the controller and instrument loop in which they are contained. These links are specifically addressed in the procedure to be closed during the restoration process. Testing involves performing an instrument loop calibration and string check (IP). This IP verifies correct output of the manual loader at 0, 25, 50, 75, and 100% of range and verifies proper response of other components in the instrument loop. Since the controller will be programmed to function as a manual loader, there are no tuning requirements, and since the IP verifies proper valve response, no other testing is required. All testing will be completed prior to entering Mode 4 to ensure proper system operation before returning it to service.

The equipment affected by this modification is not required to be operable or perform a safety function when this procedure is used. This procedure creates no new failure modes, and failure of the manual loader is not evaluated in the FSAR. No operating parameters or safety limits will be changed, and setpoints will be returned to their normal operating value before return to service.

In conclusion, there will be no Unreviewed Safety Questions created by this procedure.

PT/1/A/4200/01A, Retype, Changes 0 to 10 Incorporated

PT/1/A/4200/01A, Containment Integrated Leak Rate Test (ILRT) is performed in mode 5 or No Mode to satisfy Surveillance Requirements 4.6.1.2, 4.6.1.6 and

4.6.1.7. The test is outlined in FSAR section 6.2.6. This is essentially the same test as outlined in chapter 14 of the FSAR for the preoperational Type A test. There are some differences in the test alignments for some penetrations due to post operation system alignment needs. The procedure allows for Decay Heat Removal (ND) to remain in operation for core cooling. Additionally, the structural integrity test (test pressure at 110 - 115 percent of design pressure) is not performed. As stated in the FSAR and Tech. Specs., the test is performed in accordance with Appendix J of 10CFR50 and the provisions of ANSI N45.4-1972 or the Mass Point method.

The test is performed by sealing the containment vessel and pumping in air to achieve a pressure of 14.88 psig to 15.0 psig. A four hour stabilization period is then entered. If stable conditions exist after at least four hours, the test is started. A test pressure of 14.88 - 15.0 psig is used to ensure that the containment pressure remains greater than 14.68 psig (Pa) for the duration of the test. After completion of the test, a known leak is imposed on the containment vessel to verify the accuracy of the leak rate calculations. Given satisfactory results, the containment vessel is depressurized and the test concluded.

Due to the increased air density inside containment when pressurized for the test, the containment fire detectors may go into alarm. A fire watch will be established by using the Resistance Temperature Devices (RTDs) as the means of fire detection. These RTDs are located in lower containment, upper containment and the ice condenser, thus providing adequate coverage of the entire containment.

The Containment Integrated Leak Rate Test is performed in Mode 5 or No Mode. Containment closure/integrity is not required during the performance of this test since core alterations or mid-loop operations would not be conducted during ILRT. The test alignment for penetrations is written such that closure is maintained since it is possible that penetrations (other than M371) could still be in the test alignment prior to or following ILRT when closure/integrity is required.

Due to the increased air density inside containment during ILRT, it is necessary to decrease the speed of the ice condenser (NF) air handling unit (AHU) fans. It will also be necessary to increase the capacity of the thermal overloads for the NF AHU fan motors. The pressure increase inside containment will result in a doubling of the air density. Since fan power is directly proportional to fan rpm cubed times the density, the power required by the fan motor will double. Therefore, with this reduced speed and doubled air density, the required fan motor power will be 2 times the normal value. This will be accounted for by increasing the capacity of the thermal overloads for the fan motors to 2 times their current capacity. Due to the length of time required to install these modifications, installation could start prior to entering Mode 5. These fans are not safety related, are not required for safe shutdown of the unit, and have no Tech. Specs. directly related to them; therefore, operability is not a concern. They do, however, maintain the ice condenser at its required temperature. The ice condenser temperature is

monitored at least every 12 hours as required by Tech. Spec. surveillance requirement 4.6.5.1.a; therefore, any problem with ice condenser temperature would be discovered and corrective action taken. This test will be logged

into the shift supervisor's logbook prior to implementing these modifications and will remain logged in until the test is complete and the modifications removed.

During the ILRT, the Reactor Cooling System Vent for Low Temperature Overpressure Protection (LTOP) must be maintained. Since the procedure requires venting the cold leg accumulators which supply the safety related motive force for the PORV's during LTOP, it is necessary to either gag open two NC PORVs, or at least one NC Code Safety Valve must be removed to establish a 4.5 in² vent path for the Reactor Coolant System. This procedure includes a step to ensure that one of these methods is used.

A depressurization rig will be installed at a spare NF Penetration (M371) which will be used to depressurize containment following completion of the Integrated Leak Rate Test. This assembly is not seismically qualified to meet requirements for containment integrity or containment closure. Thus, this assembly can be installed when containment integrity/closure is not required, and must be removed prior to any activity/condition which requires containment integrity/closure. Special provisions are included in the procedure to ensure that the depressurization rig is removed prior to Mode 6, and procedure steps are provided to ensure that the blind flanges have been reinstalled on M371 prior to Mode 4 entry.

Also, a Containment Air Release and Addition (VQ) modification will be performed on check valve 1VQ12 to allow flow to pass in the other direction. This will be an alternate depressurization path, if needed, following the completion of the Integrated Leak Rate Test. The modifications discussed above are installed and removed by the direction of this procedure which includes independent verification.

The relief valve, 1VY34, on Penetration M346 will not be used. Penetration M346 will now be used for a depressurization path following the completion of the Integrated Leak Rate Test.

Therefore, the probability and consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR will not be increased. The possibility of an accident or malfunction of equipment important to safety not previously evaluated in the FSAR will not be created. The margin of safety in the basis to Tech. Specs. will not be reduced. No unreviewed safety question is created.

TN/1/A/1242/00/01A, Initial Issue

Nuclear Station Modification (NSM) CN-11242, Rev. 0, will modify the control circuit wiring on 1CA002, 004, 006, 1FW001A, 032B, 1KC003A, 018B, 228B, 230A, 305B, 315B, 320A, 332B, 333A, 338B, 424B, 425A, 1ND032A, 065B, 1NF233B, 1RF389B, 447B, and 457B to provide "limit actuated" torque switch bypass contacts which can be adjusted independently of indications or interlocks and provide data to complete Motor Operated Valve (MOV) testing of the valves. The MOV testing information included in the NSM will supersede the old torque switch setting values and replace them with thrust values. The settings are selected, set, and maintained correctly to accommodate the maximum differential pressure expected on the valve during both normal and abnormal

events within the design basis. The new thrust values ensure the valve will operate during normal and abnormal events by setting limitations on Total Thrust, Differential Pressure (DP) Thrust, and Packing Load.

This procedure will control work being performed on Auxiliary Feedwater (CA) valve 1CA002. Instrument and Electrical (IAE) will perform all work at the valve. IAE will rewire the rotors, set up the switch rotors, verify add-on-pak switch setup, and perform MOV testing of the valve.

This procedure may be implemented with Unit 1 in Mode 4, 5, 6, or No Mode. Valve 1CA002 is the CA pump suction from hotwell isolation valve. The CA system is not required to be operable in these modes. Prior to returning the valve to service, a functional verification will be performed to verify valve operability.

An unreviewed safety question does not exist.

TN/1/A/1242/00/02A, Initial Issue

Nuclear Station Modification (NSM) CN-11242, Rev. 0, will modify the control circuit wiring on 1CA002, 004 [illegible] 1FW[illegible] 032B, 1KC003A, 018B, 228B, 230A, 305B, 315B, 320A, 332B, 333A, [illegible] 42[illegible] 1ND032A, 065B, 1NF233B, 1RF389B, 447B, and 457B to pro[illegible]d" torque switch bypass contacts which can be adjusted [illegible] indications or interlocks and provide data to complete Motor Opera[illegible] valve (MOV) testing of the valves. The MOV testing information included in the NSM will supersede the old torque switch setting values and replace them with thrust values. The settings are selected, set, and maintained correctly to accommodate the maximum differential pressure expected on the valve during both normal and abnormal events within the design basis. The new thrust values ensure the valve will operate during normal and abnormal events by setting limitations on Total Thrust, Differential Pressure (DP) Thrust, and Packing Load.

This procedure will control work being performed on Auxiliary Feedwater (CA) valve 1CA004. Instrument and Electrical (IAE) will perform all work at the valve. IAE will rewire the rotors, set up the switch rotors, verify add-on-pak switch setup, and perform MOV testing of the valve.

This procedure may be implemented with Unit 1 in Modes 4, 5, 6, or No Mode. Valve 1CA004 is the CA pump suction from Upper Surge Tank Header valve. The CA system is not required to be operable in these modes. Prior to returning the valve to service, a functional verification will be performed to verify valve operability.

An unreviewed safety question does not exist.

TN/1/A/1242/00/03A, Initial Issue

Nuclear Station Modification (NSM) CN-11242, Rev. 0, will modify the control circuit wiring on 1CA002, 004, 006, 1FW001A, 032B, 1KC003A, 018B, 228B, 230A, 305B, 315B, 320A, 332B, 333A, 338B, 424B, 425A, 1ND032A, 065B, 1NF233B, 1RF389B, 447B, and 457B to provide "limit actuated" torque switch bypass contacts which can be adjusted independently of indications or interlocks and provide data to complete Motor Operated Valve (MOV) testing of the valves. The MOV testing information included in the NSM will supersede the old torque switch setting values and replace them with thrust values. The settings are selected, set, and maintained correctly to accommodate the maximum differential pressure expected on the valve during both normal and abnormal events within the design basis. The new thrust values ensure the valve will operate during normal and abnormal events by setting limitations on Total Thrust, Differential Pressure (DP) Thrust, and Packing Load.

This procedure will control work being performed on Auxiliary Feedwater (CA) valve 1CA006. Instrument and Electrical (IAE) will perform all work at the valve. IAE will rewire the rotors, set up the switch rotors, verify add-on-pak switch setup, and perform MOV testing of the valve.

This procedure may be implemented with Unit 1 in Mode 4, 5, 6, or No Mode. Valve 1CA006 is the supply from CA Condensate Storage Isolation valve. The CA System is not required to be operable in these modes. Prior to returning the valve to service, a functional verification will be performed to verify valve operability.

An unreviewed safety question does not exist.

TN/1/A/2679/CE/01A, Initial Issue

This procedure provides guidance for the implementation of Exempt Change CE-2679, Work Unit 01A. This modification replaces manual loader 1NVML2940 and controllers 1NVSS5571 and 1NVSS5651 with Moore Products 352B Single Loop Digital Controllers. The manual loader provides for manual control of valve 1NV294 and is adjusted to regulate charging flow from the Centrifugal Charging Pumps. Controller 1NVSS5571 operates valve 1NV148, and controls letdown pressure downstream of the Letdown Heat Exchanger to prevent flashing. Controller 1NVSS5651 operates valve 1NV309, and controls seal water injection flow to the Reactor Coolant (NC) pumps. All are located on Auxiliary Shutdown Panel (ASP) A and provide a secondary means of control for these valves. The new controllers will be programmed to function as single loop controllers for 1NVSS5571 and 1NVSS5651, and as a manual loader for 1NVML2940, thereby providing the same function as the existing manual loader and controllers.

All work in this procedure will be performed with Unit 1 in Modes 4, 5, 6, or No Mode. Control of letdown, charging, or NC seal water flow from 1ASPA is not required in these modes. The panel on which the manual loader and controllers are located will not be adversely affected by the work performed in this procedure, and no other equipment will be affected. Electrical isolations for this procedure involve opening/closing sliding links and will only affect the controllers and instrument loops they are contained in. These links are specifically addressed in the procedure to be closed during the

restoration process. Testing involves performing an instrument loop calibration and string check (IP) and verifying the ability of the controllers to maintain specific setpoints with satisfactory process response. The IPs verify correct output of the manual loader, and of the controllers, when placed in manual mode, at 0, 25, 50, 75, and 100% of range, and verify proper response of other components in the instrument loop. Since the manual loader has no tuning requirements, and the IPs verify proper valve response, no other testing is required. For the controllers, values of flow/pressure expected during normal operating conditions will be used as setpoints to verify the tuning constants provide adequate control. All testing will be completed prior to entering Mode 3 to ensure proper system operation before returning it to service.

The equipment affected by this modification is not required to be operable or perform a safety function when this procedure is used. This procedure creates no new failure modes, and failure of the loader or controllers is not evaluated in the FSAR. No operating parameters or safety limits will be changed. Setpoints will be returned to their normal operating value before return to service.

In conclusion, there will be no Unreviewed Safety Questions created by this procedure, and it will not require any changes to the Technical Specifications or FSAR.

TN/1/A/2679/CE/02A, Initial Issue

This procedure provides guidance for the implementation of Exempt Change CE-2679, Work Unit 02A. This modification replaces manual loader 1NVML2941 and controller 1NVSS5652 with Moore Products 352B Single Loop Digital Controllers. Manual Loader 1NVML1241 provides for manual control of valve 1NV124B, which is adjusted to maintain excess letdown pressure to prevent exceeding the allowable back pressure on the Reactor Coolant (NC) pump #1 seals. Controller 1NVSS5652 operates valve 1NV309 and controls seal water injection flow to the NC pumps. Both are located on Auxiliary Shutdown Panel (ASP) B and provide a secondary means of control for these valves. The new controllers will be programmed to function as a single loop controller for 1NVSS5652 and as a manual loader for 1NVML1241, thereby providing the same function as the existing manual loader and controller.

All work in this procedure will be performed with Unit 1 in Modes 4, 5, 6, or No Mode. Control of letdown or NC seal water flow from 1ASPB is not required in these modes. The panel on which the manual loader and controllers are located will not be adversely affected by the work performed in this procedure, and no other equipment will be affected. Electrical isolations for this procedure involve opening/closing sliding links and will only affect the controllers and instrument loops they are contained in. These links are specifically addressed in the procedure to be closed during the restoration process. Testing involves performing an instrument loop calibration and string check (IP) and verifying the ability of the controllers to maintain specific setpoints with satisfactory process response. The IPs verify correct output of the manual loader, and of the controller, when placed in manual mode, at 0, 25, 50, 75, and 100% of range, and verify proper response of other components in the instrument loop. Since the manual loader has no tuning

requirements, and the IPs verify proper valve response, no other testing is required. For the controllers, values of flow expected during normal operating conditions will be used as setpoints to verify the tuning constants provide adequate control. All testing will be completed prior to entering Mode 3 to ensure proper system operation before returning it to service.

The equipment affected by this modification is not required to be operable or perform a safety function when this procedure is used. This procedure creates no new failure modes, and failure of the loader or controllers is not evaluated in the FSAR. No operating parameters or safety limits will be changed, and setpoints will be returned to their normal operating value before return to service.

In conclusion, there will be no Unreviewed Safety Questions created by this procedure, and it will not require any changes to the Technical Specifications or FSAR.

TN/1/A/3282/CE/01A, Initial Issue

This procedure provides guidance for the implementation of Exempt Change CE-3282, Work Unit 01A. This modification provides manual throttling capability for Safety Injection (NI) valve 1NI173A. This will be accomplished by adding a third position to the power disconnect switch, located on 1MC11, which will defeat the seal-in circuitry for the M/O and M/C circuits. This will allow 1NI173A to be positioned to any intermediate position by depressing the OPEN or CLOSE pushbutton, then releasing the pushbutton when the valve reaches the desired position. This capability will only be available when the disconnect switch is placed in the throttle (new) position.

All work in this procedure will be performed with Unit 1 in Modes 5, 6, or No Mode and Decay Heat Removal (ND) Train A out of service. No plant equipment other than the valve will be affected. The changes described in this procedure involve only electrical wiring changes to the valve control circuits; no changes to the valve or valve actuator will be performed. Electrical isolations involve red tagging specific breakers open. Testing involves stroking the valve full open and closed to verify present valve operating characteristics are maintained. The steps contained in Periodic Test PT/1/A/4200/18 which verifies power removal for 1NI173A will be performed to verify the ability to remove power to the valve actuator, and hence compliance with Technical Specification 3/4.5.2, has not been degraded. Since the purpose of this modification is to provide throttling capability, the valve will be placed in the throttle mode and the control board pushbutton used to position the valve to various intermediate positions between full open and full closed. All testing will be completed prior to entering Mode 4 to ensure proper system operation before returning it to service.

This procedure will be performed when ND Train A is not required to be operable or perform a safety function in Modes 5, 6, or No Mode. This procedure creates no new failure modes, and the failure analysis presented in the FSAR for this valve will not be affected. No operating parameters, safety limits, or setpoints will be changed.

In conclusion, there will be no Unreviewed Safety Questions created by this procedure, and it will not require any changes to the Technical Specifications or FSAR.

TN/1/A/3282/CE/02A, Initial Issue

This procedure provides guidance for the implementation of Exempt Change CE-3282, Work Unit 02A. This modification provides manual throttling capability Safety Injection (NI) valve 1NI178B. This will be accomplished by adding a third position to the power disconnect switch, located on 1MC11, which will defeat the seal-in circuitry for the M/O and M/C circuits. This will allow 1NI178B to be positioned to any intermediate position by depressing the OPEN or CLOSE pushbutton, then releasing the pushbutton when the valve reaches the desired position. This capability will only be available when the disconnect switch is placed in the throttle (new) position.

All work in this procedure will be performed with Unit 1 in Modes 5, 6, or No Mode and Decay Heat Removal (ND) Train B out of service. No plant equipment other than the valve will be affected. The changes described in this procedure involve only electrical wiring changes to the valve control circuits; no changes to the valve or valve actuator will be performed. Electrical isolations involve red tagging specific breakers open. Testing involves stroking the valve full open and closed to verify present valve operating characteristics are maintained. The steps contained in Periodic Test PT/1/A/4200/18 which verifies power removal for 1NI178B will be performed to verify the ability to remove power to the valve actuator, and hence compliance with Technical Specification 3/4.5.2, has not been degraded. Since the purpose of this modification is to provide throttling capability, the valve will be placed in the throttle mode and the control board pushbutton used to position the valve to various intermediate positions between full open and full closed. All testing will be completed prior to entering Mode 4 to ensure proper system operation before returning it to service.

This procedure will be performed when ND Train B is not required to be operable or perform a safety function in Modes 5, 6, or No Mode. This procedure creates no new failure modes, and the failure analysis presented in the FSAR for this valve will not be affected. No operating parameters, safety limits, or setpoints will be changed.

In conclusion, there will be no Unreviewed Safety Questions created by this procedure and it will not require any changes to the Technical Specifications or FSAR.

TN/2/B/0615/00/01A, Initial Issue

TN/2/B/0615/00/01A replaces the level transmitter, 2CSLT5840, and adds two current alarm modules for the Unit 2 Upper Surge Tanks in accordance with Nuclear Station Modification (NSM) CN-20615. NSM CN-20615 was written to provide more accurate level indication for the Unit 2 Upper Surge Tanks. The Upper Surge Tanks are now being operated in an overflow condition to insure that the station complies with Technical Specification 3/4.7.1.5. This technical specification requires that the Condensate Storage System be

operable with a water volume of at least 225,000 gallons. The Control Room recorder 2CSCR5840 is constantly off scale due to the surge tanks being in a overflow condition.

NSM CN-20615 replaces level transmitter 2CSLT5840 and reroutes its impulse line. The low pressure input to 2CSLT5840 will be attached to the Upper Surge Dome vent line; the high pressure input will be attached to the bottom of the Upper Surge Tank. CN-20615 replaces pressure switches 2CSPS5840, 5841, and 5842 with two current alarms. NSM CN-20615 also adds level gauge 2CSLG5970. Level gauge 2CSLG5970 provides the station with level indication while the new level transmitter and electrical connections are being installed.

2CSLG5970 will be installed under TN/2/B/0615/00/01B. Condensate water volume will be verified at the Upper Surge Tanks by 2CSLG5970. TN/2/B/0615/00/01A will install the remaining portion of NSM CN-20615. TN/2/B/0615/00/01A will install the new level transmitter, current alarm modules, and will rescale pin #2 of control room chart recorder 2CSCR5840. Control Room indication of condensate water volume and its associated alarms will be restored by TN/2/B/0615/00/01A.

The transmitter and current alarms have no control function. The work covered by this procedure involves components described in Technical Specification 3/4.7.1.5. The Technical Specification requires verification of the Condensate Storage System water volume. This requirement will be satisfied by 2CSLG5970 until TN/2/B/0615/00/01A is completed. TN/2/B/0615/00/01A will make operable the alarms referenced the FSAR section 9.2.6.4. The implementation of this procedure will not create an unreviewed safety question.

TN/2/B/0615/00/01B, Initial Issue

TN/2/B/0615/00/01B adds a visual level indicator to the Unit 2 Upper Surge Tanks in accordance with Nuclear Station Modification (NSM) CN-20615. NSM CN-20615 was written to provide more accurate level indication for the Unit 2 Upper Surge Tanks. The Upper Surge Tanks are now being operated in an overflow condition to insure that the station complies with Technical Specification 3/4.7.1.5. This technical specification requires that the Condensate Storage System be operable with a water volume of at least 225,000 gallons. The Control Room recorder 2CSCR5840 is constantly off scale due to the surge tanks being in a overflow condition.

NSM CN-20615 replaces level transmitter 2CSLT5840 and reroutes its impulse line. The low pressure input to 2CSLT5840 will be attached to the Upper Surge Dome vent line; the high pressure input will be attached to the bottom of the Upper Surge Tank. CN-20615 replaces pressure switches 2CSPS5840, 5841, and 5842 with two current alarms. NSM CN-20615 also adds level gauge 2CSLG5970. Level gauge 2CSLG5970 will provide the station with level indication while the new level transmitter and electrical connections are being installed.

TN/2/B/0615/00/01B installs 2CSLG5970. The existing level transmitter, 2CSLT5840, will not be operational while level gauge is being installed. The new transmitter for 2CSLT5840 will be installed by another TN procedure.

The transmitter and current alarms have no control function. The work covered by this procedure involves components described in Technical Specification 3/4.7.1.5. The Technical Specification requires verification of the Condensate Storage System water volume. This specification allows level indication to be inoperable for seven days if the standby nuclear service water pond, as backup supply to the auxiliary feedwater pumps, is operable. The duration of this modification should be two days. Temporary level indication can be utilized if needed.

MP/0/A/7600/23A, Retype, Changes 0 to 1 Incorporated

This procedure provides a method for disassembly, inspection, reassembly, and corrective maintenance for BIF Butterfly (BF) Valves (HW ONLY).

The purpose of this evaluation is to describe changes made to MP/0/A/7600/23A due to a valve modification on selected BIF BF valves. The modification consists of the following:

1 - Machining the valve body for installation of a mechanically retained seat,

2 - A re-designed two piece seat to provide more reliable sealing,

3 - Addition of internal mechanical stops to prevent the possibility of the valve disc from rotating through the seat.

The modification should improve the valves' ability to reliably isolate plant equipment. Appropriate sections were added to the valve procedure to allow the disassembly, inspection, and reassembly of the modified valves.

The purpose of this procedure is to correct and improve the performance of these valves within their original design requirements and specifications. The modification has been reviewed as a part of Nuclear Station Modifications CN-11223 and CN-20599, and Exempt Changes CE3171 and CE3172. The procedure will maintain the valves in the as designed condition and will not significantly change the component.

No unreviewed safety questions are involved.

TN/1/A/1242/00/04A, Initial Issue

Nuclear Station Modification (NSM) CN-11242, Rev. 0, will modify the control circuit wiring on 1CA002, 004, 006, 1FW001A, 032B, 1KC003A, 018B, 228B, 230A, 305B, 315B, 320A, 332B, 333A, 338B, 424B, 425A, 1ND032A, 065B, 1NF233B, 1RF389B, 447B, and 457B to provide "limit actuated" torque switch bypass contacts which can be adjusted independently of indications or interlocks and provide data to complete Motor Operated Valve (MOV) testing of the valves.

This procedure will control work being performed on Refueling Water (FW) valve 1FW001A. Instrument and Electrical (IAE) personnel will perform all work at the valve. IAE will rewire the rotors, set up the switch rotors, and verify add-on-pak switch setup.

This procedure will be implemented with Unit 1 in Modes 5, 6, or No Mode. Valve 1FW001A is the Refueling Water Loop Isolation valve. Valve 1FW001A closes on receipt of an SS-K signal and provides redundant isolation of the class E (non-safety) portion of the Refueling Water system from the class B (safety) portion, which is used as a part of the ECCS. (In the event of an emergency, it isolates the suction of the Refueling Water Pump from the Refueling Water Storage Tank.) This valve is a normally closed, electric motor operated (EMO) seismic category I valve. This valve may be inoperable in Modes 5, 6, and no mode. Prior to returning the valve to service, a functional verification and retest will be performed to verify valve operability.

An unreviewed safety question does not exist.

TN/1/A/1242/00/05A, Initial Issue

Nuclear Station Modification (NSM) CN-11242, Rev. 0, will modify the control circuit wiring on 1CA002, 004, 006, 1FW001A, 032B, 1KC003A, 018B, 228B, 230A, 305B, 315B, 320A, 332B, 333A, 338B, 424B, 425A, 1ND032A, 065B, 1NF233B, 1RF389B, 447B, and 457B to provide "limit actuated" torque switch bypass contacts which can be adjusted independently of indications or interlocks and provide data to complete Motor Operated Valve (MOV) testing of the valves.

This procedure will control work being performed on Refueling Water (FW) valve 1FW032B. Instrument and Electrical (IAE) personnel will perform all work at the valve. IAE will rewire the rotors, set up the switch rotors, and verify add-on-pak switch setup.

This procedure will be implemented with Unit 1 in Modes 5, 6, or No Mode. Valve 1FW032B closes on receipt of an SS-K signal and provides redundant isolation of the class E (non-safety) portion of the Refueling Water system from the class B (safety) portion, which is used as a part of the ECCS. (In the event of an emergency, it isolates the suction of the Refueling Water Pump from the Refueling Water Storage Tank.) This valve is a normally closed, electric motor operated (EMO) seismic category I valve. This valve may be inoperable in Modes 5, 6, and no mode. Valve 1FW032B is the Refueling Water Loop Isolation valve. Prior to returning the valve to service, a functional verification and retest will be performed to verify valve operability.

An unreviewed safety question does not exist.

TN/1/A/1242/00/06A, Initial Issue

Nuclear Station Modification (NSM) CN-11242, Rev. 0, will modify the control circuit wiring on 1CA002, 004, 006, 1FW001A, 032B, 1KC003A, 018B, 228B, 230A, 305B, 315B, 320A, 332B, 333A, 338B, 424B, 425A, 1ND032A, 065B, 1NF233B, 1RF389B, 447B, and 457B to provide "limit actuated" torque switch bypass contacts which can be adjusted independently of indications or interlocks and provide data to complete Motor Operated Valve (MOV) testing of the valves.

This procedure will control work being performed on Residual Heat Removal (ND) valve 1ND032A. Instrument and Electrical (IAE) personnel will perform all work at the valve. IAE wi. set up the switch rotors and verify add-on-pak switch setup.

This procedure will be implemented with Unit 1 in Modes 5, 6, or No Mode. Valve 1ND032A is the ND Train 1A Hot Leg Injection Return Isolation valve. Performance of this TN will require cycling the valve. This valve is a normally open, motor operated gate valve, located in the cross-connecting piping downstream of the ND Heat Exchangers. This valve is used to align the ND system for the recirculation phases following a loss of coolant accident (LOCA). It is manually controlled from the Control Room. This valve has no interlocks. It is manually closed when establishing cold leg recirculation following a LOCA. This valve is not required to be operable during Modes 5, 6, and no mode because the Emergency Core Cooling System (ECCS) is not required to be operable in these modes. It is not required for Residual Heat Removal. Prior to returning the valve to service, a functional verification and retest will be performed to verify valve operability.

An unreviewed safety question does not exist.

TN/1/A/1242/00/07, Initial Issue

Nuclear Station Modification (NSM) CN-11242, Rev. 0, will modify the control circuit wiring on 1CA002, 004, 006, 1FW001A, 032B, 1KC003A, 018B, 228B, 230A, 305B, 315B, 320A, 332B, 333A, 338B, 424B, 425A, 1ND032A, 065B, 1NF233B, 1RF389B, 447B, and 457B to provide "limit actuated" torque switch bypass contacts which can be adjusted independently of indications or interlocks and provide data to complete Motor Operated Valve (MOV) testing of the valves.

This procedure will control work being performed on Residual Heat Removal (ND) valve 1ND065B. Instrument and Electrical (IAE) personnel will perform all work at the valve. IAE set up the switch rotors and verify add-on-pak switch setup.

This procedure will be implemented with Unit 1 in Modes 5, 6 or No Mode. Valve 1ND065B is the ND Train 1B Hot Leg Injection Return Isolation valve. Performance of this TN will require cycling the valve. This valve is a normally open, motor operated gate valve, located in the cross-connecting piping downstream of the ND Heat Exchangers. This valve is used to align the ND system for the recirculation phases following a loss of coolant accident (LOCA). It is manually controlled from the Control Room. This valve has no interlocks. It is manually closed when establishing cold leg recirculation following a LOCA. This valve is not required to be operable during Modes 5, 6, and no mode because the Emergency Core Cooling System (ECCS) is not required to be operable in these modes. It is not required for Residual Heat Removal. Prior to returning the valve to service, a functional verification and retest will be performed to verify valve operability.

An unreviewed safety question does not exist.

TN/1/A/1242/00/08A, Initial Issue

Nuclear Station Modification (NSM) CN-11242, Rev. 0, will modify the control circuit wiring on 1CA002, 004, 006, 1FW001A, 032B, 1KC003A, 018B, 228B, 230A, 305B, 315B, 320A, 332B, 333A, 338B, 424B, 425A, 1ND032A, 065B, 1NF233B, 1RF389B, 447B, and 457B to provide "limit actuated" torque switch bypass contacts which can be adjusted independently of indications or interlocks and provide data to complete Motor Operated Valve (MOV) testing of the valves. The MOV testing information included in the NSM will supersede the old torque switch setting values and replace them with thrust values. The settings are selected, set, and maintained correctly to accommodate the maximum differential pressure expected on the valve during both normal and abnormal events within the design basis. The new thrust values ensure the valve will operate during normal and abnormal events by setting limitations on Total Thrust, Differential Pressure (DP) Thrust, and Packing Load.

This procedure will control work being performed on Ice Condenser Refrigeration (NF) valve 1NF233B. Instrument and Electrical (IAE) personnel will perform all work at the valve. IAE will rewire the rotors, set up the switch rotors, and verify add-on-pak switch setup, and perform MOV testing of the valve.

This procedure will be implemented with Unit 1 in Modes 5, 6, or No Mode. Valve 1NF233B is the Air Handling Unit Glycol Return Inside Containment Isolation valve. Performing the work discussed in this TN will require removing this valve from service. This valve closes in response to an ST (Containment Isolation, Phase A) signal to isolate the glycol piping inside the containment from the external refrigeration system. This isolates the glycol supply to the ice condenser air handling units. Containment Isolation Valves are not required to be operable in Modes 5, 6, or No Mode. The ice condenser is not required to be operable in Modes 5, 6, or No Mode. Prior to returning the valve to service, a functional verification and retest will be performed to verify valve operability.

An unreviewed safety question does not exist.

TN/1/A/1242/00/12A, Initial Issue

Nuclear Station Modification (NSM) CN-11242, Rev. 0, will modify the control circuit wiring on 1CA002, 004, 006, 1FW001A, 032B, 1KC003A, 018B, 228B, 230A, 305B, 315B, 320A, 332B, 333A, 338B, 424B, 425A, 1ND032A, 065B, 1NF233B, 1RF389B, 447B, and 457B to provide "limit actuated" torque switch bypass contacts which can be adjusted independently of indications or interlocks and provide data to complete Motor Operated Valve (MOV) testing of the valves. The MOV testing information included in the NSM will supersede the old torque switch setting values and replace them with thrust values. The settings are selected, set, and maintained correctly to accommodate the maximum differential pressure expected on the valve during both normal and abnormal events within the design basis. The new thrust values ensure the valve will operate during normal and abnormal events by setting limitations on Total Thrust, Differential Pressure (DP) Thrust, and Packing Load.

This procedure will control work being performed on Component Cooling (KC) valve 1KC003A. Instrument Electrical (IAE) personnel will perform all work at the valve. IAE will rewire the rotors, set up the switch rotors, verify add-on-pak switch setup, and perform MOV testing of the valve.

This procedure will be implemented with Unit 1 in Modes 5, 6, or No Mode. Valve 1KC003A is the Reactor Building Non-Essential Return Header Isolation valve. Performing the modification on valve 1KC003A will require de-energizing the valve. The valve will fail "as-is". Later, the valve will be cycled to verify the new switch settings, and perform stroke time testing. Per Tech. Spec. 3.7.3, Component Cooling is not required to be operable in the modes in which work will be performed. Prior to returning the valve to service, a functional verification and retest will be performed to verify valve operability.

An unreviewed safety question does not exist.

TN/1/A/1242/00/13A, Initial Issue

Nuclear Station Modification (NSM) CN-11242, Rev. 0, will modify the control circuit wiring on 1CA002, 004, 006, 1FW001A, 032B, 1KC003A, 018B, 228B, 230A, 305B, 315B, 320A, 332B, 333A, 338B, 424B, 425A, 1ND032A, 065B, 1NF233B, 1RF389B, 447B, and 457B to provide "limit actuated" torque switch bypass contacts which can be adjusted independently of indications or interlocks and provide data to complete Motor Operated Valve (MOV) testing of the valves. The MOV testing information included in the NSM will supersede the old torque switch setting values and replace them with thrust values. The settings are selected, set, and maintained correctly to accommodate the maximum differential pressure expected on the valve during both normal and abnormal events within the design basis. The new thrust values ensure the valve will operate during normal and abnormal events by setting limitations on Total Thrust, Differential Pressure (DP) Thrust, and Packing Load.

This procedure will control work being performed on Component Cooling (KC) valve 1KC018B. Instrument and Electrical (IAE) personnel will perform all work at the valve. IAE will rewire the rotors, set up the switch rotors, and verify add-on-pak switch setup, and perform MOV testing of the valve.

This procedure will be implemented with Unit 1 in Modes 5, 6, or No Mode. Valve 1KC018B is the Reactor Building Non-Essential Return Header Isolation valve. Performing the modification on valve 1KC018B will require de-energizing the valve. The valve will fail "as-is". Later, the valve will be cycled to verify the new switch settings, and perform stroke time testing. Per Tech. Spec. 3.7.3, Component Cooling is not required to be operable in Modes 5, 6, or No Mode. Prior to returning the valve to service, a functional verification and retest will be performed to verify valve operability.

An unreviewed safety question does not exist.

TN/1/A/1242/00/14A, Initial Issue

Nuclear Station Modification (NSM) CN-11242, Rev. 0, will modify the control circuit wiring on 1CA002, 004, 006, 1FW001A, 032B, 1KC003A, 018B, 228B, 230A, 305B, 315B, 320A, 332B, 333A, 338B, 424B, 425A, 1ND032A, 065B, 1NF233B, 1RF389B, 447B, and 457B to provide "limit actuated" torque switch bypass contacts which can be adjusted independently of indications or interlocks and provide data to complete Motor Operated Valve (MOV) testing of the valves. The MOV testing information included in the NSM will supersede the old torque switch setting values and replace them with thrust values. The settings are selected, set, and maintained correctly to accommodate the maximum differential pressure expected on the valve during both normal and abnormal events within the design basis. The new thrust values ensure the valve will operate during normal and abnormal events by setting limitations on Total Thrust, Differential Pressure (DP) Thrust, and Packing Load.

This procedure will control work being performed on Component Cooling (KC) valve 1KC228B. Instrument and Electrical (IAE) personnel will perform all work at the valve. IAE will rewire the rotors, set up the switch rotors, and verify add-on-pak switch setup, and perform MOV testing of the valve.

This procedure will be implemented with Unit 1 in Modes 5, 6, or No Mode. Valve 1KC228B is the Train 1B Supply to Reactor Building Non-Essential Header Isolation valve. The isolations required to perform this work will cause 1KC228B to fail "as is." In addition, 1KC002B, 18B, 53B, and 228B will close (due to a simulated low-low KC surge tank level from the isolations.) A number of indications relating to these components will be affected. Tech. Spec. 3.7.3 only requires KC to be operable in Modes 1-4. KC is designed for operation during all phases of plant operation. The non-essential header can be supplied from either KC train. The isolations will close the train B valves for the Auxiliary Building and Reactor Building Non- ssential Headers. Thus, the isolations required by this procedure may remove parts of one train of KC from service. However, any loads required to be served during the outage could be supplied by the redundant train. Instructions are given in the procedure to inform the Unit Supervisor of the effects of the isolations required to perform this TN. Operations has the responsibility to ensure KC configuration control. Prior to returning the valve to service, a functional verification and retest will be performed to verify valve operability.

An unreviewed safety question does not exist.

TN/1/A/1242/00/15A, Initial Issue

Nuclear Station Modification (NSM) CN-11242, Rev. 0, will modify the control circuit wiring on 1CA002, 004, 006, 1FW001A, 032B, 1KC003A, 018B, 228B, 230A, 305B, 315B, 320A, 332B, 333A, 338B, 424B, 425A, 1ND032A, 065B, 1NF233B, 1RF389B, 447B, and 457B to provide "limit actuated" torque switch bypass contacts which can be adjusted independently of indications or interlocks and provide data to complete Motor Operated Valve (MOV) testing of the valves. The MOV testing information included in the NSM will supersede the old torque switch setting values and replace them with thrust values. The settings are selected, set, and maintained correctly to accommodate the maximum differential pressure expected on the valve during both normal and abnormal

events within the design basis. The new thrust values ensure the valve will operate during normal and abnormal events by setting limitations on Total Thrust, Differential Pressure (DP) Thrust, and Packing Load.

This procedure will control work being performed on Component Cooling (KC) valve 1KC230A. Instrument Electrical (IAE) personnel will perform all work at the valve. IAE will rewire the rotors, set up the switch rotors, verify add-on-pak switch setup, and perform MOV testing of the valve.

This valve is the Train 1A Supply to Reactor Building Non-Essential Header Isolation Valve. The isolations required to perform this work will cause 1KC230A to fail "as is." In addition, 1KC001A, 003A, 053A, and 230A will close (due to a simulated low-low KC surge tank level from the isolations.) A number of indications relating to these components will be affected.

This work will be performed with Unit 1 in Modes 5, 6, or No Mode. Tech. Spec. 3.7.3 only requires KC to be operable in Modes 1-4. KC is designed for operation during all phases of plant operation. The non-essential header can be supplied from either KC train. The isolations will close the train A valves for the Auxiliary Building and Reactor Building Non-essential Headers. Thus, the isolations required by this procedure may remove parts of one train of KC from service. However, any loads required to be served during the outage could be supplied by the redundant train. Instructions are given in the procedure to inform the Unit Supervisor of the effects of the isolations required to perform this TN. Operations has the responsibility to ensure KC configuration control. Prior to returning the valve to service, a functional verification and retest will be performed to verify valve operability.

An unreviewed safety question does not exist.

TN/1/A/1242/00/16A, Initial Issue

Nuclear Station Modification (NSM) CN-11242, Rev. 0, will modify the control circuit wiring on 1CA002, 004, 006, 1FW001A, 032B, 1KC003A, 018B, 228B, 230A, 305B, 315B, 320A, 332B, 333A, 338B, 424B, 425A, 1ND032A, 065B, 1NF233B, 1RF389B, 447B, and 457B to provide "limit actuated" torque switch bypass contacts which can be adjusted independently of indications or interlocks and provide data to complete Motor Operated Valve (MOV) testing of the valves. The MOV testing information included in the NSM will supersede the old torque switch setting values and replace them with thrust values. The settings are selected, set, and maintained correctly to accommodate the maximum differential pressure expected on the valve during both normal and abnormal events within the design basis. The new thrust values ensure the valve will operate during normal and abnormal events by setting limitations on Total Thrust, Differential Pressure (DP) Thrust, and Packing Load.

This procedure will control work being performed on Component Cooling (KC) valve 1KC305B. Instrument and Electrical (IAE) personnel will perform all work at the valve. IAE will rewire the rotors, set up the switch rotors, verify add-on-pak switch setup, and perform MOV testing of the valve.

This procedure will be implemented with Unit 1 in Modes 5, 6, or No Mode. Valve 1KC305B is the Excess Letdown Heat Exchanger Supply Header Containment

Isolation (Outside) Valve. The isolations required to perform this work will cause 1KC305B to fail "as is."

Tech. Spec. 3.7.3 only requires KC to be operable in Modes 1-4. This valve is a containment isolation valve. Tech. Spec. 3.6.3 concerning containment isolation valves is only applicable in Modes 1-4. The Excess Letdown Heat Exchanger is not used when the unit is being cooled by the Decay Heat Removal (ND) system, as it would be in modes 5 or 6. Prior to returning the valve to service, a functional verification and retest will be performed to verify valve operability.

An unreviewed safety question does not exist.

TN/1/A/1242/00/17A, Initial Issue

Nuclear Station Modification (NSM) CN-11242, Rev. 0, will modify the control circuit wiring on 1CA002, 004, 006, 1FW001A, 032B, 1KC003A, 018B, 228B, 230A, 305B, 315B, 320A, 332B, 333A, 338B, 424B, 425A, 1ND032A, 065B, 1NF233B, 1RF389B, 447B, and 457B to provide "limit actuated" torque switch bypass contacts which can be adjusted independently of indications or interlocks and provide data to complete Motor Operated Valve (MOV) testing of the valves. The MOV testing information included in the NSM will supersede the old torque switch setting values and replace them with thrust values. The settings are selected, set, and maintained correctly to accommodate the maximum differential pressure expected on the valve during both normal and abnormal events within the design basis. The new thrust values ensure the valve will operate during normal and abnormal events by setting limitations on Total Thrust, Differential Pressure (DP) Thrust, and Packing Load.

This procedure will control work being performed on Component Cooling (KC) valve 1KC315B. Instrument and Electrical (IAE) personnel will perform all work at the valve. IAE will rewire the rotors, set up the switch rotors, verify add-on-pak switch setup, and perform MOV testing of the valve.

This TN performs modification work on valve 1KC315B. This valve is the Excess Letdown Heat Exchanger Return Header Containment Isolation (Outside) Valve. The isolations required to perform this work will cause 1KC315B to fail "as is."

This work will be performed with Unit 1 in Modes 5, 6, or No Mode. Tech. Spec. 3.7.3 only requires KC to be operable in Modes 1-4. This valve is a containment isolation valve. Tech. Spec. 3.6.3 concerning containment isolation valves is only applicable in Modes 1-4. The Excess Letdown Heat Exchanger is not used when the unit is being cooled by the Decay Heat Removal (ND) system, as it would be in modes 5 or 6. Prior to returning the valve to service, a functional verification and retest will be performed to verify valve operability.

An unreviewed safety question does not exist.

TN/1/A/1242/00/18A, Initial Issue

Nuclear Station Modification (NSM) CN-11242, Rev. 0, will modify the control circuit wiring on 1CA002, 004, 006, 1FW001A, 032B, 1KC003A, 018B, 228B, 230A, 305B, 315B, 320A, 332B, 333A, 338B, 424B, 425A, 1ND032A, 065B, 1NF233B, 1RF389B, 447B, and 457B to provide "limit actuated" torque switch bypass contacts which can be adjusted independently of indications or interlocks and provide data to complete Motor Operated Valve (MOV) testing of the valves. The MOV testing information included in the NSM will supersede the old torque switch setting values and replace them with thrust values. The settings are selected, set, and maintained correctly to accommodate the maximum differential pressure expected on the valve during both normal and abnormal events within the design basis. The new thrust values ensure the valve will operate during normal and abnormal events by setting limitations on Total Thrust, Differential Pressure (DP) Thrust, and Packing Load.

This procedure will control work being performed on Component Cooling (KC) valve 1KC320A. Instrument and Electrical (IAE) personnel will perform all work at the valve. IAE will rewire the rotors, set up the switch rotors, verify add-on-pak switch setup, and perform MOV testing of the valve.

This TN performs modification work on valve 1KC320A. This valve is the Reactor Coolant Drain Tank Heat Exchanger Cooling Water Supply Containment Isolation (Outside) Valve. The isolations required to perform this work will cause 1KC320A to fail "as is."

This work will be performed with Unit 1 in Modes 5, 6, or No Mode. Tech. Spec. 3.7.3 only requires KC to be operable in Modes 1-4. This valve is a containment isolation valve. Tech. Spec. 3.6.3 concerning containment isolation valves is only applicable in Modes 1-4. The Reactor Coolant Drain Tank Heat Exchanger does not have a heat load in modes 5 or 6. The normal position for 1KC320A is open (providing flow.) Even if this valve were to be cycled closed, there would be no safety concern in the applicable modes. Prior to returning the valve to service, a functional verification and retest will be performed to verify valve operability.

An unreviewed safety question does not exist.

TN/1/A/1242/00/19A, Initial Issue

Nuclear Station Modification (NSM) CN-11242, Rev. 0, will modify the control circuit wiring on 1CA002, 004, 006, 1FW001A, 032B, 1KC003A, 018B, 228B, 230A, 305B, 315B, 320A, 332B, 333A, 338B, 424B, 425A, 1ND032A, 065B, 1NF233B, 1RF389B, 447B, and 457B to provide "limit actuated" torque switch bypass contacts which can be adjusted independently of indications or interlocks and provide data to complete Motor Operated Valve (MOV) testing of the valves. The MOV testing information included in the NSM will supersede the old torque switch setting values and replace them with thrust values. The settings are selected, set, and maintained correctly to accommodate the maximum differential pressure expected on the valve during both normal and abnormal events within the design basis. The new thrust values ensure the valve will operate during normal and abnormal events by setting limitations on Total Thrust, Differential Pressure (DP) Thrust, and Packing Load.

This procedure will control work being performed on Component Cooling (KC) valve 1KC332B. Instrument and Electrical (IAE) personnel will perform all

work at the valve. IAE will rewire the rotors, set up the switch rotors, verify add-on-pak switch setup, and perform MOV testing of the valve.

This TN performs modification work on valve 1KC332B. This valve is the Reactor Coolant Drain Tank Heat Exchanger Cooling Water Return Containment Isolation (Inside) Valve. The isolations required to perform this work will cause 1KC332B to fail "as is

This work will be performed with Unit 1 in Modes 5, 6, or No Mode. Tech. Spec. 3.7.3 only requires KC to be operable in Modes 1-4. This valve is a containment isolation valve. Tech. Spec. 3.6.3 concerning containment isolation valves is only applicable in Modes 1-4. The Reactor Coolant Drain Tank Heat Exchanger does not have a heat load in modes 5 or 6. The normal position for 1KC332B is open (providing flow.) Even if this valve were to be cycled closed, there would be no safety concern in the applicable modes. Prior to returning the valve to service, a functional verification and retest will be performed to verify valve operability.

An unreviewed safety question does not exist.

TN/1/A/1242/00/20A, Initial Issue

Nuclear Station Modification (NSM) CN-11242, Rev. 0, will modify the control circuit wiring on 1CA002, 004, 00?, 1FW001A, 0?2B, 1KC003A, 018B, 228B, 230A, 305B, 315B, 320A, 332B, 333A, 33??, 424B, 425A, 1ND032A, 065B, 1NF233B, 1RF389B, 447B, and 457B to provide "limit actuated" torque switch bypass contacts which can be adjusted independently of indications or interlocks and provide data to complete Motor Operated Valve (MOV) testing of the valves. The MOV testing information included in the NSM will supersede the old torque switch setting values and replace them with thrust values. The settings are selected, set, and maintained correctly to accommodate the maximum differential pressure expected on the valve during both normal and abnormal events within the design basis. The new thrust values ensure the valve will operate during normal and abnormal events by setting limitations on Total Thrust, Differential Pressure (DP) Thrust, and Packing Load.

This procedure will control work being performed on Component Cooling (KC) valve 1KC333A. Instrument and Electrical (IAE) personnel will perform all work at the valve. IAE will rewire the rotors, set up the switch rotors, verify add-on-pak switch setup, and perform MOV testing of the valve.

This TN performs modification work on valve 1KC333A. This valve is the Reactor Coolant Drain Tank Heat Exchanger Cooling Water Return Containment Isolation (Outside) Valve. The isolations required to perform this work will cause 1KC333A to fail "as is."

This work will be performed with Unit 1 in Modes 5, 6, or No Mode. Tech. Spec. 3.7.3 only requires KC to be operable in Modes 1-4. This valve is a containment isolation valve. Tech. Spec. 3.6.3 concerning containment isolation valves is only applicable in Modes 1-4. The Reactor Coolant Drain Tank Heat Exchanger does not have a heat load in modes 5 or 6. The normal position for 1KC333A is open (providing flow.) Even if this valve were to be cycled closed, there would be no safety concern in the applicable modes.

Prior to returning the valve to service, a functional verification and retest will be performed to verify valve operability.

An unreviewed safety question does not exist.

TN/1/A/1242/00/21A, Initial Issue

Nuclear Station Modification (NSM) CN-11242, Rev. 0, will modify the control circuit wiring on 1CA002, 004, 006, 1FW001A, 032B, 1KC003A, 018B, 228B, 230A, 305B, 315B, 320A, 332B, 333A, 338B, 424B, 425A, 1ND032A, 065B, 1NF233B, 1RF389B, 447B, and 457B to provide "limit actuated" torque switch bypass contacts which can be adjusted independently of indications or interlocks and provide data to complete Motor Operated Valve (MOV) testing of the valves. The MOV testing information included in the NSM will supersede the old torque switch setting values and replace them with thrust values. The settings are selected, set, and maintained correctly to accommodate the maximum differential pressure expected on the valve during both normal and abnormal events within the design basis. The new thrust values ensure the valve will operate during normal and abnormal events by setting limitations on Total Thrust, Differential Pressure (DP) Thrust, and Packing Load.

This procedure will control work being performed on Component Cooling (KC) valve 1KC338B. Instrument and Electrical (IAE) personnel will perform all work at the valve. IAE will rewire the rotors, set up the switch rotors, verify add-on-pak switch setup, and perform MOV testing of the valve.

This TN performs modification work on valve 1KC338B. This valve is the Reactor Coolant Pumps' Supply Header Containment Isolation (Outside) Valve. This valve serves the reactor vessel support coolers and reactor coolant pumps. The isolations required to perform this work will cause 1KC338B to fail "as is."

This work will be performed with Unit 1 in Modes 5, 6, or No Mode with the Reactor Coolant Pumps off. Tech. Spec. 3.7.3 only requires KC to be operable in Modes 1-4. This valve is a containment isolation valve. Tech. Spec. 3.6.3 concerning containment isolation valves is only applicable in Modes 1-4. The normal position for 1KC338B is open (providing flow.) Even if this valve were to be cycled closed, there would be no safety concern since the reactor coolant pumps will be off when this TN is performed. Prior to returning the valve to service, a functional verification and retest will be performed to verify valve operability.

An unreviewed safety question does not exist.

TN/1/A/1242/00/22A, Initial Issue

Nuclear Station Modification (NSM) CN-11242, Rev. 0, will modify the control circuit wiring on 1CA002, 004, 006, 1FW001A, 032B, 1KC003A, 018B, 228B, 230A, 305B, 315B, 320A, 332B, 333A, 338B, 424B, 425A, 1ND032A, 065B, 1NF233B, 1RF389B, 447B, and 457B to provide "limit actuated" torque switch bypass contacts which can be adjusted independently of indications or interlocks and provide data to complete Motor Operated Valve (MOV) testing of the valves.

The MOV testing information included in the NSM will supersede the old torque switch setting values and replace them with thrust values. The settings are selected, set, and maintained correctly to accommodate the maximum differential pressure expected on the valve during both normal and abnormal events within the design basis. The new thrust values ensure the valve will operate during normal and abnormal events by setting limitations on Total Thrust, Differential Pressure (DP) Thrust, and Packing Load.

This procedure will control work being performed on Component Cooling (KC) valve 1KC424B. Instrument and Electrical (IAE) personnel will perform all work at the valve. IAE will rewire the rotors, set up the switch rotors, verify add-on-pak switch setup, and perform MOV testing of the valve.

This TN performs modification work on valve 1KC424B. This valve is the Reactor Coolant Pumps' Return Header Containment Isolation (Inside) Valve. This valve serves the reactor vessel support coolers and reactor coolant pumps. The isolations required to perform this work will cause 1KC424B to fail "as is."

This work will be performed with Unit 1 in Modes 5, 6, or No Mode with the Reactor Coolant Pumps off. Tech. Spec. 3.7.3 only requires KC to be operable in Modes 1-4. This valve is a containment isolation valve. Tech. Spec. 3.6.3 concerning containment isolation valves is only applicable in Modes 1-4. The normal position for 1KC424B is open (providing flow. Even if this valve were to be cycled closed, there would be no safety concern since the reactor coolant pumps will be off when this TN is performed. Prior to returning the valve to service, a functional verification and retest will be performed to verify valve operability.

An unreviewed safety question does not exist.

TN/1/A/1242/00/23A, Initial Issue

Nuclear Station Modification (NSM) CN-11242, Rev. 0, will modify the control circuit wiring on 1CA002, 004, 006, 1FW001A, 032B, 1KC003A, 018B, 228B, 230A, 305B, 315B, 320A, 332B, 333A, 338B, 424B, 425A, 1ND032A, 065B, 1NF233B, 1RF389B, 447B, and 457B to provide "limit actuated" torque switch bypass contacts which can be adjusted independently of indications or interlocks and provide data to complete Motor Operated Valve (MOV) testing of the valves. The MOV testing information included in the NSM will supersede the old torque switch setting values and replace them with thrust values. The settings are selected, set, and maintained correctly to accommodate the maximum differential pressure expected on the valve during both normal and abnormal events within the design basis. The new thrust values ensure the valve will operate during normal and abnormal events by setting limitations on Total Thrust, Differential Pressure (DP) Thrust, and Packing Load.

This procedure will control work being performed on Component Cooling (KC) valve 1KC425A. Instrument and Electrical personnel (IAE) will perform all work at the valve. IAE will rewire the rotors, set up the switch rotors, verify add-on-pak switch setup, and perform MOV testing of the valve.

This TN performs modification work on valve 1KC425A. This valve is the Reactor Coolant Pumps' Return Header Containment Isolation (Outside) Valve. This valve serves the reactor vessel support coolers and reactor coolant pumps. The isolations required to perform this work will cause 1KC425A to fail "as is."

This work will be performed with Unit 1 in Modes 5, 6, or No Mode with the Reactor Coolant Pumps off. Tech. Spec. 3.7.3 only requires KC to be operable in Modes 1-4. This valve is a containment isolation valve. Tech. Spec. 3.6.3 concerning containment isolation valves is only applicable in Modes 1-4. The normal position for 1KC425A is open (providing flow.) Even if this valve were to be cycled closed, there would be no safety concern since the reactor coolant pumps will be off when this TN is performed. Prior to returning the valve to service, a functional verification and retest will be performed to verify valve operability.

An unreviewed safety question does not exist.

PT/2/A/4200/26, Change #20

This change adds steps to close the manual isolation valves for Residual Heat Removal (ND) containment spray prior to testing Containment Spray (NS) valves 2NS-38B and 2NS-43A. It was discovered that opening 2NS-38B and 2NS-43A while at power with the manual valves open could degrade ND system injection flow to less than that assumed for FSAR LOCA analysis. Steps have been added to record "AS FOUND" position of the respective manual valve, close the valve, and return the valve to "AS FOUND" position after testing is complete.

The test previously contained steps to verify ND discharge pressure less than 56 psig by reading a gauge located on 522' elevation. This ensured that pressure was not sufficient to force water out of the NS Spray nozzle during the test. This pressure verification is still necessary with the manual valve closed. If pressure were high enough (approximately 79 psig), water could be trapped above the check valve in that line. This could cause water to be discharged when the manual valve is opened at test completion. The wording has been changed to allow verification of less than 56 psig from Control Room gauge 2NDP5080 for train B and 2NDP5090 for train A.

This change also adds steps to test 2NS-38B and 2NS-43A using Containment Pressure Control System (CPCS) interlocks. Previously, the valves were given a containment pressure permissive and then opened and closed from the control room. This change documents that, given a pressure permissive, the valves are allowed to be opened from the control room, and that the valves automatically close when the permissive is removed. This documentation will allow the CPCS Periodic Test procedure to be deleted as it is being incorporated into other tests. Valve stroke timing will be performed using these interlocks.

Independent verification of manual valve position ensures that the valves are closed for test and returned to the as found position at test completion. No unreviewed safety question is created.

PT/1/A/4200/26, Change #36

The purpose of this test is to satisfy the requirements of Section XI, Subsection IWV of ASME Boiler and Pressure Code with regard to the measurement of valve stroke time, valve operability and valve position indicator verification as required by Catawba IWV Submittal.

Containment Spray (NS) Valves 1NS38B and 1NS43A are being changed to cold shutdown testing. McGuire LER 369/90-22 states that opening these valves while at power could degrade Decay Heat Removal (ND) system operation in the event of a Large Break Loss Of Coolant Accident (LBLOCA) in which injection flow would be diverted from both trains of the ND system. Therefore, these valves will only be stroke time tested during Modes 5, 6, or no mode. By keeping these valves closed during power operation, and therefore not degrading the ND system, the consequences of an accident is decreased.

Valves 1NS42 and 1NS47 are being deleted. When closed, these valves isolate ND from the NS Spray Header. With this test now being performed in Modes 5, 6, or no mode, there is no longer a need for ND to be isolated from the NS Spray Header. By deleting these valves, there is no longer a need for operations support in lower containment, thus reducing Catawba's Personnel Radiation Exposure.

By placing Solid State Protection System (SSPS) 1SSPSA and 1SSPSB in test position, an inadvertent safety system actuation is defeated, and therefore spray into upper containment will be prevented.

No unreviewed safety question is created.

PT/2/A/4200/26, Change #23

The purpose of this test is to satisfy the requirements of Section XI, Subsection IWV of ASME Boiler and Pressure Code with regard to the measurement of valve stroke time, valve operability and valve position indicator verification as required by Catawba IWV Submittal.

Containment Spray (NS) Valves 2NS38B and 2NS43A are being changed to cold shutdown testing. McGuire LER 369/90-22 states that opening these valves while at power could degrade Decay Heat Removal (ND) system operation in the event of a Large Break Loss Of Coolant Accident (LBLOCA) in which injection flow would be diverted from both trains of the ND system. Therefore, these valves will only be stroke time tested during Modes 5, 6, or no mode. By keeping these valves closed during power operation, and therefore not degrading the ND system, the consequences of an accident is decreased.

Valves 2NS42 and 2NS47 are being deleted. When closed, these valves isolate ND from the NS Spray Header. With this test now being performed in Modes 5, 6, or no mode, there is no longer a need for ND to be isolated from the NS Spray Header. By deleting these valves, there is no longer a need for operations support in lower containment, thus reducing Catawba's Personnel Radiation Exposure.

By placing Solid State Protection System (SSPS) 2SSPSA and 2SSPSB in test position, an inadvertent safety system actuation is defeated, and therefore spray into upper containment will be prevented.

No unreviewed safety question is created.

TN/1/A/1163/00/02A, Initial Issue

Implementation procedure TN/1/A/1163/00/02A replaces the present Unit 1 Turbine Controls System with a new Digital Turbine Controls System in accordance with Nuclear Station Modification (NSM) CN-11163, Rev. 0. This control system incorporates circuitry and equipment to control the operation of the Main Turbine Generator.

Work activities within this procedure will require that turbine control functions be disabled. Disabling the control functions requires the turbine to be shut down and Unit 1 to be in Modes 4, 5, 6, or No-Mode.

The present turbine controls system is non-safety. However, this system and other systems affected by the implementation of this procedure are Tech. Spec. related and will require certain actions. Because certain Reactor Trip signals are related to the turbine control system, mode requirements are stated in accordance with the associated Tech. Spec. All the turbine overspeed protection will be disabled, and therefore this procedure will be worked when turbine overspeed protection is not required per Tech. Specs.

Another system affected by this procedure is Control Room Ventilation. The Control Room Ventilation system will be affected by pulling cables through Control Room firestops. This will require the use of a Compensatory action form to provide specific instructions for opening and closing Control Room firestops such that Control Room Ventilation will remain operable. A Compensatory action form will also be required for opening the Control Room door for extended periods of time. This door will be opened to vent fumes related to welding being performed per this procedure.

All Main Control Board devices affected per this procedure will have stickers placed on them to ensure that the Operator at the Controls (OATC) is aware of the affected Control Board indications. Isolations in this procedure are made to ensure that both equipment important to safety and personnel safety are maintained. Station control procedures will be used to perform the specific work activities as stated in the procedure.

Upon completion of the implementation of this procedure, all the Main Turbine System input devices will be calibrated and tested. Verification of all the Turbine Trip interlocks will be performed. Devices relocated to implement this procedure will be functionally verified. Control Room Ventilation will be retested to ensure operability in accordance with Tech. Specs. Operations' Start-up testing, Control Valve testing, Stop Valve testing, and Turbine Overspeed testing will ensure proper operation of the new system. This testing will be documented on a separate work request for this NSM.

The implementation of this procedure in accordance with the design for NSM CN-11163, Rev. 0 will not create an unreviewed safety question.

IP/0/B/3314/30M, Change #3

This change deleted the output blocking for OEMF43A and OEMF43B, Control Room Ventilation (VC) Intake Monitors. The control function (closing of the intake dampers) was removed per Nuclear Station Modification (NSM) CN-50422. The Technical Specification change was made to Tech. Spec. 3.3.3.1, and the need for an FSAR change was identified in calculation CNC-1503.13-00-0369.

The change does not affect any of the Chapter 15 analyses. No fission product barriers or source term evaluations are affected by the change. The Post Loss of Coolant Accident control room dose calculations are likewise unaffected. Failure will not create a situation which has not been considered in the FSAR. Neither VC nor OEMF43A and B interact with other safety equipment. The ability of the VC system to maintain control room and control room area temperature and pressure is not affected. No setpoints, design limits or operating parameters have been affected by this change.

In conclusion, there are no Unreviewed Safety Questions associated with this change. No FSAR revisions are required beyond those required by CNC-1503.13-00-0369.

IP/0/B/3314/15, Change #34

This change deleted the output blocking for OEMF43A and OEMF43B, Control Room Ventilation (VC) Intake Monitors. The control function (closing of the intake dampers) was removed per Nuclear Station Modification (NSM) CN-50422. A Technical Specification change was made to Tech. Spec. 3.3.3.1, and the need for an FSAR change was identified in calculation CNC-1503.13-00-0369.

The change does not affect any of the Chapter 15 analyses. No fission product barriers or source term evaluations are affected by the change. The Post Loss of Coolant Accident control room dose calculations are likewise unaffected. Failure will not create a situation which has not been considered in the FSAR. Neither VC nor OEMF43A and B interact with other safety equipment. The ability of the VC system to maintain control room and control room area temperature and pressure is not affected. No new safety related equipment has been added. No setpoints, design limits or operating parameters have been affected by this change.

In conclusion, there are no Unreviewed Safety Questions associated with this change. No FSAR revisions are required beyond those required by CNC-1503.13-00-0369.

TN/1/A/1168/00/02A, Initial Issue

This procedure will perform numerous wiring changes to the plant as part of the outage work associated with Nuclear Station Modification (NSM) CN-11168, Rev. 0. NSM CN-11168, Rev. 0 replaces the existing Feedwater Control System for Unit 1 with a new Digital Feedwater Control System (DFCS).

This procedure will provide guidelines for the activities required to install the new Digital Feedwater Control System. Due to the number of changes

associated with this modification, the Feedwater System (CF) will be rendered inoperable while the NSM is being implemented. Therefore, this new control system will be installed during a refueling outage for Unit 1. Specifically, this procedure will be performed while Unit 1 is in Modes 5, 6, or No Mode.

The present Feedwater Control System, as well as the new Digital Feedwater Control System, is non-safety. However, there are several Tech. Spec. items associated with the Feedwater System. The mode requirements which have been established for implementation of this modification are such that the Feedwater System will not be required by the Tech. Specs.

There is a substantial quantity of work to be performed in the Control Room area for this NSM. The Control Room Ventilation System (VC) will be affected by pulling cables through Control Room firestops. A Compensatory Action Form will be used to provide specific instructions for opening and closing firestops such that Control Room Ventilation will remain operable and positive pressure will be maintained. Personnel will be prepared to immediately seal the penetrations should the need arise. All Main Control Board chart recorders and gauges affected by this procedure will have out of service stickers placed on them to ensure that the operator at the controls is aware of the affected indications. The out of service stickers will be removed at various times during implementation of the modification to allow the Control Room to have required indications operable as needed.

Isolations in this procedure are made to ensure that both equipment important to plant safety and personnel safety are maintained. Station control procedures will be used to perform the specific work activities as stated in the procedure.

Upon completion of the implementation of this procedure, all of the Digital Feedwater Control System input devices will be calibrated and tested. Also, all devices in the Process Control Cabinets which are relocated to implement this modification will be functionally verified. Calibrations will be performed on all valves and transmitters affected by this procedure to ensure proper operation.

The implementation of this procedure will not create an unreviewed safety question.

MP/0/A/7600/130, Initial Issue

This procedure is to perform corrective maintenance on Atwood and Morrill Bolted Bonnet Swing Check Valves with bolted hanger brackets. This procedure is applicable to valves [illegible] n various system applications (presently the only application where [illegible] alves are installed is in the Liquid Radioactive Waste system).

This procedure has been reviewed against approved vendor manuals, design documents, and station procedures to ensure that the corrective maintenance controlled by this procedure will return the valve to as-built/as-designed condition. Therefore, no unreviewed safety question exists.

MP/0/A/7600/129, Initial Issue

This procedure is to perform corrective maintenance on Atwood and Morrill 4" Pressure Seal Check Valves. This procedure is only applicable to valves used in the auxiliary feedwater (CA) system.

This procedure has been reviewed against approved vendor manuals, design documents, and station procedures to ensure that the corrective maintenance controlled by this procedure will return the valve to as-built/as-designed condition. These actions will ensure valve compliance with FSAR accident analysis. Therefore, no unreviewed safety question exists.

MP/0/A/7600/38, Change #3

This procedure is for performing corrective maintenance. NRC Information Notice No. 90-03 identified the potential for the lower disc arm to fail (i.e. crack) causing separation of the disc from the lower arm. This change requires that a PT Examination be performed on the lower arm to identify surface cracks that may lead to failure.

The added PT examination is an additional inspection of parts already receiving a visual inspection. Warranted by the information provided in IN 90-03, this additional inspection in the procedure will improve upon the reliability of low pressure Borg-Warner swing check valves. A representative sample of valves will be inspected using this procedure in response to IN 90-03. This will insure that the possibility, consequences, and probability of an accident remains unchanged. Therefore, no unreviewed safety questions are created.

PM/IG - 114 Deletion

This safety evaluation is for the deletion of PM/IG - 114. Procedure number MP/0/A/7300/14 has been written to control ventilation equipment bearing inspection and lubrication and will be used from now on in place of PM/IG - 114.

A full safety evaluation was performed on MP/0/A/7300/14 before it was approved. MP/0/A/7300/14 was written based on approved vendor manuals, design documents, and station procedures to ensure that the actions controlled will return the fan bearings to their as designed as built condition. Since the fan bearings will be returned to their as designed conditions, the possibility, consequences, or probability of a malfunction will be reduced. Therefore, no unreviewed safety question is created by the deletion of PM/IG - 114 or the use of MP/0/A/7300/14 in its place.

MP/0/A/7150/17 Retype, Changes 0 to 4 Incorporated

Tech. Specs. 3.1.2.1, 3.1.2.3 and 3.1.2.4 may be affected by this procedure. Operations has the responsibility and the procedures for compliance with these Tech. Specs. Maintenance will be performed on this gear drive when Tech. Specs. allow, per Operation's procedures. This rewrite will clarify and

assure that maintenance activities will return the gear drive to as-designed conditions.

The changes made by this rewrite have been reviewed against approved vendor manuals, design documents, and station procedures to ensure that the corrective maintenance controlled by this procedure will return the gear drive to as-built/as-designed conditions. These actions will ensure the gear drive's compliance with FSAR accident analysis. Since the gear drive will be returned to as-designed conditions, the possibility, consequences or probability of a malfunction will be reduced. Therefore, no unreviewed safety questions exist.

MP/0/A/7150/04, Retype, Changes 0 to 10 Incorporated

This re-write of the maintenance procedure for Component Cooling Water (KC) Pumps includes steps for the option of installing labyrinth type oil seals in place of the conventional oil lip seals. The option was requested because of housekeeping problems created by the relatively short service life (2000-3000 hours) of the lip seal. The installation option was approved according to variation notice CE-3159. Use of the labyrinth seal is expected to increase the availability and reduce the required maintenance of the KC Pumps.

The installation of the labyrinth seal does not increase the probability of previously evaluated accidents because no accident initiators are affected. The maintenance work involved requires a complete teardown of the pump, and actual installation is performed in the Mechanical Maintenance shop. Therefore, the implementing steps added to this procedure for installing the labyrinth oil seals does not create any unreviewed safety questions. Other than the oil seal steps, only minor procedure format changes were made in this procedure re-write to enhance usability.

MP/0/A/7150/16A, Change #5

Revision 5 of MP/0/A/7150/16A changes the required gap between the ends of the motor and gear unit shafts. This gap is set during pump alignment. The procedure lacked specific guidance in that no reference was made to the button machined in the gear unit shaft. This change compensates for the button so that the gap is properly set.

No significant changes to structures, systems, or components as described by the FSAR are being made by this change. No procedures, as described in the FSAR, are affected significantly by this change. This change does not involve a test or experiment as described in the FSAR. No unreviewed safety questions are created.

TN/1/B/1201/00/01A, Initial Issue

This Nuclear Station Modification (NSM) will install a set of accelerometers on each of the Lower Containment Ventilation Units (LCVUs). A total of six (6) transducers will be installed onto each of the LCVU's fan/motor. Two accelerometers will be installed on each LCVU fan and two on each LCVU fan motor. Two tach. probes will be placed near the fan and motor shaft. The

output from the accelerometers on each LCVU will be sent to one of four instrument racks located in 1ELMC0024. These accelerometers will be used to detect bearing faults and allow remote balance calculations to be performed.

This NSM is non-safety, and the installation of this NSM will not have any adverse impact on equipment used for normal plant operation. The installation process for this NSM has been established to take only one fan out of service at a time during the Shot-Peening process, so that lower containment temperature can be stabilized. After the Shot-Peening process, it may be determined that more than one unit can be taken out of service.

This procedure will install new vibration monitoring equipment and delete the old equipment in 1ELMC0024. This NSM has not changed the capability of the Lower Containment Ventilation Units from performing their intended function of maintaining lower compartment temperature within the limits of safety analysis.

No cable pulling will be allowed during containment closure or mid-loop operations. Compensatory measures for penetrating firestops were handled per station directive 2.12.7. No unreviewed safety questions were created by these activities.

TN/1/B/1201/00/02A, Initial Issue

This Nuclear Station Modification (NSM) will install a set of accelerometers on each of the Lower Containment Ventilation Units (LCVUs). A total of six (6) transducers will be installed onto each of the LCVU's fan/motor. Two accelerometers will be installed on each LCVU fan and two on each LCVU fan motor. Two tach. probes will be placed near the fan and motor shaft. The output from the accelerometers on each LCVU will be sent to one of four instrument racks located in 1ELMC0024. These accelerometers will be used to detect bearing faults and allow remote balance calculations to be performed.

This NSM is non-safety, and the installation of this NSM will not have any adverse impact on equipment used for normal plant operation. The installation process for this NSM has been established to take only one fan out of service at a time during the Shot-Peening process, so that lower containment temperature can be stabilized. After the Shot-Peening process, it may be determined that more than one unit can be taken out of service.

This procedure will install the accelerometers and the sound power phone on lower containment unit LCVU-1A. This procedure has not changed the capability of the Lower Containment Ventilation Units from performing their intended function of maintaining lower compartment temperature within the limits of safety analysis

No cable pulling will be allowed during containment closure or mid-loop operations. Compensatory measures for penetrating firestops were handled per station directive 2.12.7. No unreviewed safety questions were created by these activities.

TN/1/B/1201/00/03A, Initial Issue

This Nuclear Station Modification (NSM) will install a set of accelerometers on each of the Lower Containment Ventilation Units (LCVUs). A total of six (6) transducers will be installed onto each of the LCVU's fan/motor. Two accelerometers will be installed on each LCVU fan and two on each LCVU fan motor. Two tach. probes will be placed near the fan and motor shaft. The output from the accelerometers on each LCVU will be sent to one of four instrument racks located in 1ELMC0024. These accelerometers will be used to detect bearing faults and allow remote balance calculations to be performed.

This NSM is non-safety, and the installation of this NSM will not have any adverse impact on equipment used for normal plant operation. The installation process for this NSM has been established to take only one fan out of service at a time during the Shot-Peening process, so that lower containment temperature can be stabilized. After the Shot-Peening process, it may be determined that more than one unit can be taken out of service.

This procedure will install the accelerometers and the sound power phone on lower containment unit LCVU-1B. This procedure has not changed the capability of the Lower Containment Ventilation Units from performing their intended function of maintaining lower compartment temperature within the limits of safety analysis.

No cable pulling will be allowed during containment closure or mid-loop operations. Compensatory measures for penetrating firestops were handled per station directive 2.12.7. No unreviewed safety questions were created by these activities.

TN/1/B/1201/00/04A, Initial Issue

This Nuclear Station Modification (NSM) will install a set of accelerometers on each of the Lower Containment Ventilation Units (LCVUs). A total of six (6) transducers will be installed onto each of the LCVU's fan/motor. Two accelerometers will be installed on each LCVU fan and two on each LCVU fan motor. Two tach. probes will be placed near the fan and motor shaft. The output from the accelerometers on each LCVU will be sent to one of four instrument racks located in 1ELMC0024. These accelerometers will be used to detect bearing faults and allow remote balance calculations to be performed.

This NSM is non-safety, and the installation of this NSM will not have any adverse impact on equipment used for normal plant operation. The installation process for this NSM has been established to take only one fan out of service at a time during the Shot-Peening process, so that lower containment temperature can be stabilized. After the Shot-Peening process, it may be determined that more than one unit can be taken out of service.

This procedure will install the accelerometers and the sound power phone on lower containment unit LCVU-1C. This procedure has not changed the capability of the Lower Containment Ventilation Units from performing their intended function of maintaining lower compartment temperature within the limits of safety analysis.

No cable pulling will be allowed during containment closure or mid-loop operations. Compensatory measures for penetrating firestops were handled per station directive 2.12.7. No unreviewed safety questions were created by these activities.

TN/1/B/1201/00/05A, Initial Issue

This Nuclear Station Modification (NSM) will install a set of accelerometers on each of the Lower Containment Ventilation Units (LCVUs). A total of six (6) transducers will be installed onto each of the LCVU's fan/motor. Two accelerometers will be installed on each LCVU fan and two on each LCVU fan motor. Two tach. probes will be placed near the fan and motor shaft. The output from the accelerometers on each LCVU will be sent to one of four instrument racks located in 1ELMC0024. These accelerometers will be used to detect bearing faults and allow remote balance calculations to be performed.

This NSM is non-safety, and the installation of this NSM will not have any adverse impact on equipment used for normal plant operation. The installation process for this NSM has been established to take only one fan out of service at a time during the Shot-Peening process, so that lower containment temperature can be stabilized. After the Shot-Peening process, it may be determined that more than one unit can be taken out of service.

This procedure will install the accelerometers and the sound power phone on lower containment unit LCVU-1D. This procedure has not changed the capability of the Lower Containment Ventilation Units from performing their intended function of maintaining lower compartment temperature within the limits of safety analysis.

No cable pulling will be allowed during containment closure or mid-loop operations. Compensatory measures for penetrating firestops were handled per station directive 2.12.7. No unreviewed safety questions were created by these activities.

OP/1/A/6700/01, Change #19

Data Book Table 6.5 lists the required boron concentration for 1% and 1.3% shutdown margins as a function of temperature and burnup. It is obtained from the Catawba 1 Cycle 6 Startup and Operational Report (CNNE 1553.05-00-0010) which is prepared by Duke Power Company, Design Engineering Department. As such, it is a controlled document, and all calculations in it are done by approved methods (controlled by Design Engineering).

The main purpose of table 6.5 is to provide data for use by OP/0/A/6100/06, Reactivity Balance Calculation, in calculating required boron concentration for shutdown margin at different temperatures and burnups. By maintaining this boron concentration, the accident analyses given in FSAR chapters 15.4.6, 15.1.4 and 15.1.5 remain valid. One of the initial conditions assumed in these analyses is that the boron concentration is that required for shutdown margin.

The refueling boron concentration required to maintain Keff <0.95 with all rods in is listed on the bottom of the page. This boron concentration or 2000 ppm (whichever is more restrictive) is required for Mode 6 and is also assumed in the FSAR analysis of chapter 15.4.6.

This is a data table only, so there is no possible effect on equipment important to safety. It will not increase the possibility or consequences of an accident analyzed or different from an accident analyzed in the FSAR. No unreviewed safety question is created by this change.

OP/1/A/6550/01, Change #25

This change to OP/1/A/6550/01, DIESEL GENERATOR (D/G) FUEL OIL SYSTEM OPERATION, was developed to conduct a leakage test on the Unit 1 D/G Fuel Oil tanks. It is desired to fill all four tanks within a 24 hour period. This will render both Unit 1 D/Gs inoperable because the FSAR required 24 hour settling time of the tank will not have occurred. The inoperability of both Unit 1 D/Gs while in NO MODE is not a Unit 1 concern. There are no Tech. Spec. requirements for the D/Gs while in NO MODE. This procedure does require manipulations of the Unit 2 power supplies to Control Area Ventilation and Chilled Water Systems, 1EMXG, and 2EMXH to Unit 2 power. Additionally, this places Unit 2 in an alternate Nuclear Service Water (RN) alignment with 3 RN pumps out of service.

At the end of the 24 hour settling period, the Unit 1 D/G Fuel Oil Storage Tanks will be operable and subsequently remove any inoperability placed on the Unit 1 D/Gs due to the fuel tanks.

OP/0/B/6200/16, Change #30

Temporary Station Modification (TSM) 3727MES has replaced the Turbo ice machine with a Cryo ice machine. This change gives procedural guidance in the operation of the Cryo ice machine and removed the guidance for operating the Turbo ice machine.

The Cryo ice machine is supplied with borated water using the current ice solution system. Ice produced by the low side of the machine will be routed to the permanent chute for C Flakice ice machine by an auger/chute system. During the initial 45 minutes of operation, the ice produced is below the allowable boron limit. The first four ice drops of the machine are routed to the roof to ensure the boron limits are met. The waste ice on the Auxiliary Building roof will melt and eventually end up in the conventional waste pond. The ice machine is not required to mitigate the consequences of an accident. No unreviewed safety question is judged to be created.

PT/0/A/4150/24, Change #3

The subject change alters the reference source for obtaining spent fuel pool locations of the fuel assemblies selected for examinations. The new reference source is validated for use in PT/0/A/4150/18 and is thereby appropriate for use by this procedure.

No change to the method by which the fuel assemblies are handled or examined is involved in this change. No unreviewed safety question is created.

PT/0/A/4150/26, Retype #1, Changes 1 to 2 Incorporated

PT/0/A/4150/26, Rod Control Cluster Assembly (RCCA) Ultrasonic/Eddy Current Testing, directs all activities associated with the setup of Ultrasonic/Eddy Current examination equipment, use of this equipment to examine RCCAs, and the dismantling and removal of it following completion of examinations. The test apparatus employed performs both Ultrasonic Testing (UT), which assesses external RCCA defects (e.g., vibratory and/or RCCA Guide Card induced wear marks), and Eddy Current Testing (ECT) which quantifies RCCA clad cracking internally induced by absorber to clad contact. All examination data is obtained and analyzed by approved Vendor procedure (which is incorporated as an enclosure of this procedure). Such examinations are essential to the accurate assessment of the RCCA's cladding integrity. Extensive thinning and eventual breeching of RCCA cladding would result in the loss of the neutron absorbing material contained therein. This would have significant impact on existing nuclear safety analyses.

The equipment affected by this procedure is the RCCAs themselves. All assumptions made in existing FSAR analyses concerning RCCA manipulations will be unaffected due to the fact that the RCCAs are at all times handled in the Spent Fuel Pool (SFP) with the same handling apparatus (RCCA Change Tool) as they are during normal Fuel Assembly Component shuffles. Movement of the RCCAs from their resident Fuel Assembles to the UT/ECT examination fixture is performed under approved procedure at all times.

The examination fixture itself is designed to fit into an SFP storage rack location, simulating a fuel assembly. It has been designed to be compatible with the RCCA Change Tool. The conduits through which the RCCA's rodlets are lowered and raised (to allow full length examination) are poly coated to preclude damage to external cladding surfaces. Additionally, these conduits are sized such that they are larger than the most restrictive diameter of a Fuel Assembly guide thimble to ensure smoothest possible passage of RCCA rodlets. Independent verification of proper replacement of RCCAs into their designated fuel assemblies (F/A) per PT/0/A/4150/26 shall ensure that all RCCAs will reside in proper core locations following core reloading. Examinations are to be conducted following completion of the F/A insert shuffle.

Accuracy of examination data is validated by a rigorous pre-exam benchmark using a vendor supplied RCCA with internal and external clad defects etched into it. Proper calibration of UT and ECT equipment is verified by correct identification of geometry and depth of the manufactured defects by benchmark examination. This calibration is repeated every 12 hours for the duration of examinations.

UT/ECT examinations have absolutely no effect upon latching and manipulation of the RCCAs via Rod Control System hardware. Accident analyses based upon assumptions concerning stuck rods or misoperation of the Rod Control System are therefore unaffected.

Assurance that the structural integrity of the RCCAs is not degraded by RCCA/examination fixture interface guarantees that neutron absorbing material will remain intact during subsequent power operation. The thoroughness of this nondestructive examination program will ensure that RCCAs with degraded cladding integrity are replaced before failures during power operation can result in loss of absorber and consequent impact on shutdown margin analyses.

In summary, UT/ECT examination of RCCAs is a prudent undertaking due to industry-wide problems with Guide Card induced wear on outer cladding surfaces. Control rod repositioning has been employed to more evenly distribute this wear, and the effectiveness of this program can only be assessed by such examinations. Nuclear Safety is ultimately enhanced by these examinations. No unreviewed safety question is created.

TN/1/A/2762/CE/01A, Initial Issue

Due to a wear problem on the Rod Control Cluster Assemblies (RCCAs), it is desired to reposition the fully withdrawn elevation of the RCCAs. This procedure is written to revise the thumbwheel settings for the Control Bank RCCAs. The work required by this procedure will be done in the IRE Logic Cabinet. No isolations are required by this procedure, and the changes will be made while Unit 1 is in Modes 3, 4, 5, 6 or No Mode. Functional verification will be done by both Instrument and Electrical (IAE) and Performance. The functional verification will consist of calibrations, rod drop time testing, overlap check and insertion limit checks. This testing has been discussed with the IRE System Expert and has been determined to adequately test the modification.

A revision to the Core Operating Limits Report (COLR) will be made and issued by Design Engineering. Implementing this procedure will have no effect on Tech. Specs. An unreviewed safety question will not be created during the implementation of this procedure.

TN/1/A/1107/00/03A, Initial Issue

Nuclear Station Modification (NSM) CN-11107, Rev. 0 provides an alternate recirculation loop for the Unit 1 Boric Acid Tank (BAT) in the chemical control, purification, and makeup subsystem of the Chemical and Volume Control (NV) system. A recirculation pump with isolation valves, drain valves, and instrumentation will be added for BAT recirculation. The pump will employ suction and discharge from existing BAT connections. This new recirculation loop will provide a more thorough and uniform mixing of the BAT contents than is presently provided by the existing Boric Acid Transfer pumps.

In order to perform this procedure, the Unit 1 BAT must be drained to facilitate system tie-in's. The BAT supplies boric acid solution to support unit operation. The tank is designed to store sufficient boric acid solution for a cold shutdown from full power operation. The BAT will be drained while Unit 1 is in No Mode; therefore, unit shutdown requirements will not be a concern. After completion of the BAT tie-in's, the BAT pressure boundary will be restored, and the BAT can be returned to service to provide make-up capabilities to support unit operation as required by the FSAR.

This modification also requires system tie-in's to the Boron Recycle (NB) and Liquid Waste (WL) systems. The NB tie-in will be performed using a wet-tap procedure to eliminate system down time. No NB system isolations are required, and the wet-tap will not affect NB system operation. Therefore, no additional operational concerns are imposed. The WL tie-in will be installed in a Class 'E' section of piping which supports various station equipment drains and leak-off flows. The tie-in is located at an elevation above the main drain header; therefore, installation will not impact system draining capabilities.

This procedure installs the BAT recirculation pump, recirculation piping, along with manual isolation valves and check valves, and makes the required tie-in's to the BAT, NB and WL piping. After installation, portions of the piping will be hydrostatically tested, with the remainder visually inspected at system temperature and pressure to verify piping integrity.

No unreviewed safety questions are judged to be involved or created by this procedure. No Tech. Spec. revisions are required as a result of implementation of this procedure.

TN/1/A/2677/CE/01A, Initial Issue

This procedure provides guidance for the implementation of Exempt Change CE-2677, Work Unit 01A. This modification replaces manual loaders 1CAML0360, 0400, 0480, 0520, 0560, 0600, and 0640 with Moore Products 352B Single Loop Digital Controllers. These manual loaders provide manual control of valves 1CA36, 40, 44, 48, 52, 56, 60, and 64 and are adjusted to regulate Auxiliary Feedwater (CA) flow to the Steam Generators. The new controllers will be programmed to function as manual loaders, thereby providing the same function as the existing manual loaders.

All work in this procedure will be performed with Unit 1 in modes 5, 6, or No Mode. Auxiliary Feedwater is not required during these modes. The controllers are located on Main Control Board 1MC10. This panel will not be adversely affected by the work performed in this procedure. No other equipment will be affected. Electrical isolations for this procedure involve opening/closing sliding links and will only affect the controllers and instrument loops in which they are contained. These links are specifically addressed in the procedure to be closed during the restoration process. Testing involves performing an instrument loop calibration and string check (IP). These IPs verify correct output of the manual loader at 0, 25, 50, 75 and 100% of range and verify proper response of other components in the instrument loop. Since the controllers will be programmed to function as manual loaders, there are no tuning requirements. Since the IPs verify proper valve response, no other testing is required. All testing will be completed prior to entering Mode 4 to ensure proper system operation before returning it to service.

This procedure creates no new failure modes. Failure of the manual loaders is not evaluated in the FSAR. No operating parameters or safety limits will be changed. Setpoints will be returned to their normal operating value before return to service.

In conclusion, there will be no Unreviewed Safety Questions created by this procedure, and it will not require any changes to the Technical Specifications or FSAR.

TN/1/A/2677/CE/02A, Initial Issue

This procedure provides guidance for the implementation of Exempt Change CE-2677, Work Unit 02A. This modification replaces manual loaders 1CAML0561, and 1CAML0601 with Moore Products 352B Single Loop Digital Controllers. These manual loaders provide manual control of valves 1CA56 and 1CA60 and are adjusted to regulate Auxiliary Feedwater (CA) flow to the Steam Generators 1A and 1B from CA pump 1A. Both are located on Auxiliary Shutdown Panel A and provide a secondary means of control for these valves. The new controllers will be programmed to function as manual loaders, thereby providing the same function as the existing manual loaders.

All work in this procedure will be performed with Unit 1 in modes 5, 6, or No Mode. Auxiliary Feedwater is not required during these modes. The panel on which the manual loaders are located will not be adversely affected by the work performed in this procedure. No other equipment will be affected. Electrical isolations for this procedure involve opening/closing sliding links and will only affect the loaders and instrument loops in which they are contained. These links are specifically addressed in the procedure to be closed during the restoration process. Testing involves performing an instrument loop calibration and string check (IP). These IPs verify correct output f the manual loader at 0, 25, 50, 75 and 100% of range and verify proper response of other components in the instrument loop. Since the controllers will be programmed to function as manual loaders, there are no tuning requirements. Since the IPs verify proper valve response, no other testing is required. All testing will be completed prior to entering Mode 4 to ensure proper system operation before returning it to service.

This procedure creates no new failure modes. Failure of the manual loaders is not evaluated in the FSAR. No operating parameters or safety limits will be changed. Setpoints will be returned to their normal operating value before return to service.

In conclusion, there will be no Unreviewed Safety Questions created by this procedure, and it will not require any changes to the Technical Specifications or FSAR.

TN/1/A/2677/CE/03A, Initial Issue

This procedure provides guidance for the implementation of Exempt Change CE-2677, Work Unit 03A. This modification replaces manual loaders 1CAML0401 and 1CAML0441 with Moore Products 352B Single Loop Digital Controllers. These manual loaders provide for manual control of valves 1CA40 and 1CA44 and are adjusted to regulate Auxiliary Feedwater (CA) flow to the Steam Generators 1C and 1D rom CA pump 1B. Both are located on Auxiliary Shutdown Panel B and provide a secondary means of control for these valves. The new controllers will be programmed to function as manual loaders, thereby providing the same function as the existing manual loaders.

All work in this procedure will be performed with Unit 1 in modes 5, 6, or No Mode. Auxiliary Feedwater is not required during these modes. The panel on which the manual loaders are located will not be adversely affected by the work performed in this procedure. No other equipment will be affected. Electrical isolations for this procedure involve opening/closing sliding links and will only affect the loaders and instrument loops in which they are contained. These links are specifically addressed in the procedure to be closed during the restoration process. Testing involves performing an instrument loop calibration and string check (IP). These procedures verify correct output of the manual loader at 0, 25, 50, 75 and 100% of range and verify proper response of other components in the instrument loop. Since the controllers will be programmed to function as manual loaders, there are no tuning requirements. The tests verify proper valve response. All testing will be completed prior to entering Mode 4 to ensure proper system operation before returning it to service.

Failure of the manual loaders is not evaluated in the FSAR. No new failure modes are created. No operating parameters or safety limits will be changed, and setpoints will be returned to their normal operating value before return to service.

In conclusion, there will be no Unreviewed Safety Questions created by this procedure, and it will not require any changes to the Technical Specifications or FSAR.

TN/1/A/2677/CE/04A, Initial Issue

This procedure provides guidance for the implementation of Exempt Change CE-2677, Work Unit 04A. This modification replaces manual loaders 1CAML0361, 0481, 0521, and 0641 with Moore Products 352B Single Loop Digital Controllers. These manual loaders provide manual control of valves 1CA36, 48, 52, and 64 and are adjusted to regulate Auxiliary Feedwater (CA) flow to all four Steam Generators from CA pump #1. All are located on Auxiliary Feedwater Turbine Control Panel and provide a secondary means of control for these valves. The new controllers will be programmed to function as manual loaders, thereby providing the same function as the existing manual loaders.

All work in this procedure will be performed with Unit 1 in modes 5, 6, or No Mode. Auxiliary Feedwater is not required during these modes. The panel on which the manual loaders are located will not be adversely affected by the work performed in this procedure, and no other equipment will be affected. Electrical isolations for this procedure involve opening/closing sliding links and will only affect the loaders and instrument loops in which they are contained. These links are specifically addressed in the procedure to be closed during the restoration process. Testing involves performing an instrument loop calibration and string check (IP). These procedures verify correct output of the manual loader at 0, 25, 50, 75 and 100% of range and verify proper response of other components in the instrument loop. Since the controllers will be programmed to function as manual loaders, there are no tuning requirements, and since the procedures verify proper valve response, no other testing is required. All testing will be completed prior to entering Mode 4 to ensure proper system operation before returning it to service.

Failure of the manual loaders is not evaluated in the FSAR. No operating parameters or safety limits will be changed. Setpoints will be returned to their normal operating value before return to service.

In conclusion, there will be no Unreviewed Safety Questions created by this procedure, and it will not require any changes to the Technical Specifications or FSAR.

TN/1/A/2679/CE/01A, Initial Issue

This procedure provides guidance for the implementation of Exempt Change CE-2679, Work Unit 01A. This modification replaces manual loaders 1NVML2940 and controllers 1NVSS5571 and 1NVSS5651 with Moore Products 352B Single Loop Digital Controllers. The manual loader provides for manual control of valve 1NV294 and is adjusted to regulate charging flow from the Centrifugal Charging Pumps. Controller 1NVSS5571 operates valve 1NV148, and controls letdown pressure downstream of the Letdown Heat Exchanger to prevent flashing. Controller 1NVSS5651 operates valve 1NV309, and controls seal water injection flow to the Reactor Coolant pumps. All are located on Auxiliary Shutdown Panel (ASP) A and provide a secondary means of control for these valves. The new controllers will be programmed to function as single loop controllers for 1NVSS5571 and 1NVSS5651 and as a manual loader for 1NVML2940, thereby providing the same function as the existing manual loaders and controllers.

All work in this procedure will be performed with Unit 1 in modes 4, 5, 6, or No Mode. Control of letdown, charging, or Reactor Coolant seal water flow from 1ASPA is not required in these modes. The panel on which the manual loader and controllers are located will not be adversely affected by the work performed in this procedure. No other equipment will be affected. Electrical isolations for this procedure involve opening/closing sliding links and will only affect the controllers and instrument loops in which they are contained. These links are specifically addressed in the procedure to be closed during the restoration process. Testing involves performing an instrument loop calibration and string check (IP) and verifying the ability of the controllers to maintain a specific setpoint with satisfactory process response. The IPs verify correct output of the manual loader and of the controllers when placed in manual mode, at 0, 25, 50, 75 and 100% of range and verify proper response of other components in the instrument loop. Since the manual loader has no tuning requirements, and the IPs verify proper valve response, no other testing is required. For the controllers, valves of flow/pressure expected during normal operating conditions will be used as setpoints to verify the tuning constants provide adequate control. All testing will be completed prior to entering Mode 3 to ensure proper system operation before returning it to service.

This procedure creates no new failure modes. Failure of the loaders or controllers is not evaluated in the FSAR. No operating parameters or safety limits will be changed. Setpoints will be returned to their normal operating value before return to service.

In conclusion, there will be no Unreviewed Safety Questions created by this procedure, and it will not require any changes to the Technical Specifications or FSAR.

TN/1/A/2679/CE/02A, Initial Issue

This procedure provides guidance for the implementation of Exempt Change CE-2679, Work Unit 02A. This modification replaces manual loaders 1NVML1241 and controller 1NVSS5652 with Moore Products 352B Single Loop Digital Controllers. Manual Loader 1NVML1241 provides for manual control of valve 1NV124B, which is adjusted to maintain excess letdown pressure to prevent exceeding the allowable back pressure on the Reactor Coolant pumps' (NCP's) #1 seals. Controller 1NVSS5652 operates valve 1NV309 and controls seal water injection flow to the NCPs. Both are located on Auxiliary Shutdown Panel (ASP) B and provide a secondary means of control for these valves. The new controllers will be programmed to function as single loop controllers for 1NVSS5652, and as a manual loader for 1NVML1241, thereby providing the same function as the existing manual loader and controller.

All work in this procedure will be performed with Unit 1 in modes 4, 5, 6, or No Mode. Control of excess letdown or Reactor Coolant Pump seal water flow from 1ASPB is not required in these modes. The panel on which the manual loader and controller are located will not be adversely affected by the work performed in this procedure. No other equipment will be affected. Electrical isolations for this procedure involve opening/closing sliding links and will only affect the controllers and instrument loops in which they are contained. These links are specifically addressed in the procedure to be closed during the restoration process. Testing involves performing an instrument loop calibration and string check (IP) and verifying the ability of the controllers to maintain specific setpoints with satisfactory process response. The IPs verify correct output of the manual loader, and of the controllers when placed in manual mode, at 0, 25, 50, 75 and 100% of range, and verify proper response of other components in the instrument loop. Since the manual loader has no tuning requirements, and the IPs verify proper valve response, no other testing is required. For the controllers, values of flow expected during normal operating conditions will be used as setpoints to verify the tuning constants provide adequate control. All testing will be completed prior to entering Mode 3 to ensure proper system operation before returning it to service.

This procedure creates no new failure modes. Failure of the loader or controller is not evaluated in the FSAR. No operating parameters or safety limits will be changed. Setpoints will be returned to their normal operating value before return to service.

In conclusion, there will be no Unreviewed Safety Questions created by this procedure, and it will not require any changes to the Technical Specifications or FSAR.

PT/1/A/4200/13H Retype, Changes 0 to 9 Incorporated

The purpose of this procedure is to comply with the Catawba IWV testing program requirements for operability (full stroke exercise) for those valves listed in the procedure. During performance of this test, Safety Injection (NI) pumps A and B and Chemical Volume Control (NV) centrifugal charging pumps A and B are operated in Hot Leg and Cold Leg INJECTION with suction provided by: 1) Residual Heat Removal (ND) pumps A and B; or 2) the Refueling Water

6. Section 9.0 "Sections 12.1 through 12.29" was changed to "Enclosures 13.1 through 13.29" and "Sections 13.30 and 13.31" was changed to Enclosures 13.30 and 13.31".

7. Section 11.0 "Sections 12.1 through 12.29" was changed to "Enclosures 13.1 through 13.29" and "Sections 13.30 and 13.31" was changed to Enclosures 13.30 and 13.31".

8. Section 13.0, added "13.38 Trouble Shooting Using NI pump".

9. The following changes apply to Enclosures 13.1 through 13.29.

   - Step 8.0, PREREQUISITE SYSTEM CONDITIONS was deleted.

   - Section 12.0 was renumbered. Presently the numbering does not include the prefixed number that identifies with its existing enclosure. For instance "12.6.1" of Enclosure 13.6 now reads "12.1" of Enclosure 13.6.

   - Section 12.0, step 12.1 now reads "Ensure that all prerequisites in Section 7.0 and 8.0 are satisfied."

   - Section 12.0, step 12.3 and step 12.18, valves listed on the NI test header panel were changed to reflect its proper labeling (SV was added after the NI). "1NI391" was changed to "1NISV391". NOTE: valves 1NI122B, 1NI153A, and 1NI154B are NOT listed on the NI test header panel; therefore, they do not reflect this change.

   - Section 12.0, step 12.4 was added to assure the Test coordinator that a flow path to the NI test header exists or directs the Test coordinator to Enclosure 13.38 which provides means of establishing a flow path.

10. The following changes apply to Enclosures 13.30 and 13.31.

   - The REQUIRED UNIT STATUS section was deleted. These requirements are addressed in the body of the procedure, Section 7.0.

   - Sections 8.0 and 12.0 were renumbered. Presently the numbering does not include the prefixed number that identifies with its existing enclosure.

   - Section 8.0, steps 8.1, 8.2, and 8.3 were deleted. These requirements are addressed in the body of the procedure, Section 8.0.

   - Section 12.0, step 12.1 now reads "Ensure that all prerequisites in Section 7.0 and 8.0 are satisfied."

11. On Enclosure 13.32, pages 1 through 10, "Sections 12.1 through 12.29" was changed to "Enclosures 13.1 through 13.29."

12. On Enclosure 13.33, "Section 12.30" was changed to "Enclosure 13.30."

13. On Enclosure 13.34, "Section 12.31" was changed to "Enclosure 13.31."

14. On Enclosure 13.35, "Section 12.1 through 12.29" was changed to "Enclosures 13.1 through 13.29".

15. Enclosure 13.37 was renumbered.

16. Enclosure 13.38 was added to assist the Test coordinator in reestablishing a flow path to the NI test header.

17. Previous changes 28, 29, 30, 31, and 32 have been incorporated.

The purpose of this procedure is to satisfy the Surveillance requirements of Tech. Spec. 4.4.6.2.2. The testing is done to ensure early detection of possible in-series check valve failure. The valves which 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 (LOCA) that bypasses containment.

This change will provide guidance in re-establishing a flow path to the Safety Injection (NI) test header. Historically, NI95A has failed closed repeatedly during the performance of this test. The original procedure provided no steps in re-establishing a flow path to the NI test header in the event NI95A closes. This change will allow flow to be re-established to the NI test header by operating the NI pump(s) per OP/1/A/6200/09 (Increasing Accumulator Level). Operating the NI pump(s) to increase the accumulator level is the original design of the system.

During the entire test, only one train of NI will be inoperable while re-establishing a flow path to the NI test header. Therefore, the consequences of accidents already evaluated in the FSAR remains the same. No unreviewed safety question is created.

PT/1/A/4350/02E Retype, Changes 0 to 51 Incorporated

This procedure is extremely diverse in the variety and number of surveillance requirements it fulfills. Therefore, in order to adequately address all unreviewed safety questions that may exist, this evaluation is divided into the following 5 sections:

A) Motor-Driven Auxiliary Feedwater (CA) Pumps' Auto-Starts
B) Turbine Driven CA Pump's Auto-Starts
C) Feedwater Isolations
D) Main Turbine Trips
E) ATWS/AMSAC Actuations

A) <u>Motor-Driven CA Pumps' Auto-Starts</u>
(Enclosures 13.1, 13.2, 13.5, and 13.6)

In these Enclosures, the pump under test is aligned in recirculation to the Upper Surge Tank. Various automatic start signals are simulated in order to response time test the pump, as well as to verify proper

actuation of the associated flow control valves. Accident flowrate is set up in advance by Operations to ensure that conditions exist for a valid pump response time.

Each of these Enclosures requires the Unit to be below Mode 3 (i.e., Modes 4, 5, 6, or No Mode) when the CA System is not required to be Operable. The pumps are operated well within their design limits.

B) Turbine-Driven CA Pump's Auto-Starts
(Enclosures 13.3, 13.4, 13.7, and 13.8)

In these Enclosures, CA Pump #1 is aligned in recirculation to the Upper Surge Tank. Various automatic start signals are simulated in order to response time test the pump, as well as to verify proper actuation of the associated flow control valves and steam admission valves. Accident flowrate is set up in advance by Operations to ensure that conditions exist for a valid pump response time.

Each of these Enclosures requires the secondary side Steam Generator (S/G) pressure to be above 900 psig (Modes 1, 2, or 3) in order to run CA Pump #1. Therefore, at least one motor-driven CA Pump will remain operable during the performance of any of these Enclosures. The automatic pump start signal is initiated by placing a jumper that affects no other components. The pump will be operated well within its design parameters.

C) Feedwater Isolations
(Enclosures 13.9 and 13.12)

These Enclosures will test Main Feedwater (CF) Isolation on Hi-Hi Doghouse Level and Hi-Hi S/G Level. The Hi-Hi Doghouse Level is simulated by jumper placements, while the Hi-Hi S/G Level is simulated by manipulation of Process Control Cabinet Logic Cards, and placement of one train of Solid State Protection System (SSPS) in TEST mode.

Placing the jumpers for Hi-Hi Doghouse Level causes Feedwater Isolation of valves corresponding to the appropriate Doghouse. The simulated signals cause these valves to fail to their fail safe positions, and thus do not affect the operability of any components, regardless of the mode. Process Control Cabinet modifications and placement of one train of SSPS to TEST mode cause no additional safety concerns in modes 5, 6, or No Mode (when these Enclosures are performed). The Reactor will be tripped, and the CA System will not be in operation (and is not required) during the performance of any of these Enclosures. Precautionary steps are included to ensure that the Main Feedwater System is in a configuration to prevent an undesirable increase in S/G levels. The only equipment operation is the normal stroking of valves to their fail-safe positions.

D) Main Turbine Trips
(Enclosures 13.10, 13.11, 13.13, and 13.14)

The Main Turbine will be tripped on the following signals: Loss of Both CF Pumps, Reactor Trip, Hi-Hi S/G Level, and Manual Trip. The Main Turbine is not safety related. These trips will be performed with no

steam passing through the Turbine (i.e., with the Main Steam Isolation Valves closed). There will be no transient effect on the Reactor Coolant System, and no interaction with any safety related systems. Turbine Trip capability is not required in the modes that these Enclosures will be performed.

E) <u>ATWS/AMSAC Actuation</u>
(Enclosure 13.15)

The Main Turbine will be tripped, and both Motor-Driven CA Pumps will simulate a start on the following ATWS/AMSAC signals: Loss of Both CF Pumps, and Loss of 3/4 CF Flowpaths. Since the ATWS/AMSAC System is not a system required by Technical Specifications, operability is not a concern, except for the fact that some safety related trip signals and interlocks are blocked in this Enclosure in order to verify the actuation signal is positively originated by the ATWS/AMSAC System. The valves manipulated for this Enclosure are not required for containment closure, since the S/G's are assumed to be adequate barriers to potential releases of radioactivity. If the need should arise, this Enclosure could be aborted, and all CF Valves could be closed immediately. None of these components are required in Modes 4, 5, 6, or No Mode.

No unreviewed safety questions are created by this procedure.

PT/0/A/4400/02, Initial Issue

On August 29, 1990, Problem Investigation Report (PIR) 0-C90-279 was written because it was found that testing of the fire hose stations was inadequate to ensure design flow and pressure was being delivered. The fire hose system is designed to deliver a flow of 75-100 gpm at a minimum pressure of 65 psig. It was recommended that several of the fire hose stations in the Auxiliary Building be flow tested to ensure that no degradation has occurred that would affect the fire protection system.

In order to perform this testing, a periodic test (PT/0/A/4400/02) was written. This test will involve full flow testing of several hose stations throughout the Auxiliary Building and will be performed once every five years. The water will be discharged into groundwater drainage (WZ) Sump C. For the hose stations on the 543' elevation, pressure and flowrate data will be taken. For the other elevations, a restrictive orifice will be inserted between the end of the hose and the discharge to the drain, and the pressure back at the hose station valve will be monitored. The orifice will restrict the flow to a range of 75-100 gpm through the pressure range of 60-100 psig.

All the fire hose stations that are to be tested are required by Chapter 16.9-4 of the FSAR (Selected Licensee Commitment Manual) to be operable whenever equipment that is protected by them is required to be operable. During this test, the hose station being tested will be inoperable due to its normal fire hose being disconnected and a test hose being connected between the hose station and the discharge drain. There is a Caution statement at the beginning of each test that says that the fire hose should be disconnected from the hose station for less than an hour, or some means of protection for

the areas left unprotected will be provided. Since each hose station will be inoperable for less than an hour, and since there will be someone at the hose station valve to shut it and reconnect the fire hose in case of a fire, none of the equipment protected by the hose stations will be left unprotected.

During the performance of this test, it will be necessary to keep open certain fire doors and tornado pressure doors. In order to do this, the Compensatory Action Guidelines for Plant Access Doors must be followed. This document gives the requirements for closing these doors and the time the action must be performed in.

This test does not affect any other part of the Fire Protection System. There are sufficient compensatory actions in place to return those parts of the system the test does affect to their normal conditions. None of the equipment protected by the hose stations is left unprotected. The conditions of this test do not violate the design basis assumptions as described in the FSAR. No unreviewed safety question is created.

PT/0/A/4400/08A, Initial Issue

The purpose of this procedure is to ensure that all Train A safety related components receive adequate cooling water during a faulted Engineered Safety Features (ESF) situation from the Nuclear Service Water (RN) System. This procedure does not create an unreviewed safety question.

There are two different component balances performed using this procedure. One is the full flow balance, which includes all necessary plant components, such as the Component Cooling (KC) Heat Exchangers, the Containment Spray (NS) Heat Exchangers, and the Diesel Generator Jacket Cooling Water (KD) Heat Exchangers. The other portion of the flow balance will only set flows pertaining to the pump itself, such as bearing lubrication, motor cooling, and bearing oil cooling.

During the full flow balance, the trains are first isolated from each other. Neither train is made inoperable, although the component on which maintenance was performed is considered to be inoperable. The component flows are adjusted, and throttle valves are locked once all flows are within the specified range.

Upon the cleaning of a KC and/or NS Heat Exchanger, the associated RN train will be considered to remain operable; only the isolated heat exchanger will be considered inoperable. During the post cleaning flow balance, the addressed heat exchanger will gradually be brought back into service while the key flows are monitored for their maintained operability.

During the pumphouse portion flow balance, only the pump under test is placed in service. Should the pumphouse flows fail to meet their "Allowable Flow" limits, but meet the "Minimum Allowable Flow" limits, the crossover valves are opened, and pump discharge pressure is returned to the one pump operation status. Neither train of RN is made inoperable by this procedure. Each pump is tested individually for "Allowable Flow" limits at this increased pressure.

The RN System is not degraded by this test. The RN pumps are not operated outside design conditions. The performance of the test does not make any train of RN inoperable.

PT/0/A/4400/08B, Initial Issue

The purpose of this procedure is to ensure that all Train B safety related components receive adequate cooling water during a faulted Engineered Safety Features (ESF) situation from the Nuclear Service Water (RN) System. This procedure does not create an unreviewed safety question.

There are two different component balances performed using this procedure. One is the full flow balance, which includes all necessary plant components, such as the Component Cooling (KC) Heat Exchangers, the Containment Spray (NS) Heat Exchangers, and the Diesel Generator Jacket Cooling Water (KD) Heat Exchangers. The other portion of the flow balance will only set flows pertaining to the pump itself, such as bearing lubrication, motor cooling, and bearing oil cooling.

During the full flow balance, the trains are first isolated from each other. Neither train is made inoperable, although the component on which maintenance was performed is considered to be inoperable. The component flows are adjusted, and throttle valves are locked once all flows are within the specified range.

Upon the cleaning of a KC and/or NS Heat Exchanger, the associated RN train will be considered to remain operable; only the isolated heat exchanger will be considered inoperable. During the post cleaning flow balance, the addressed heat exchanger will gradually be brought back into service while the key flows are monitored for their maintained operability.

During the pumphouse portion flow balance, only the pump under test is placed in service. Should the pumphouse flows fail to meet their "Allowable Flow" limits, but meet the "Minimum Allowable Flow" limits, the crossover valves are opened, and pump discharge pressure is returned to the one pump operation status. Neither train of RN is made inoperable by this procedure. Each pump is tested individually for "Allowable Flow" limits at this increased pressure.

The RN System is not degraded by this test. The RN pumps are not operated outside design conditions. The performance of the test does not make any train of RN inoperable.

PT/0/A/4400/22A, Change #31

The purpose of this procedure change is to allow for the Nuclear Service Water (RN) Flow Balance: Train A; the Nuclear Service Water Pump Train A Performance Test, and the RN Pump 1A Flow Versus Pressure Data Acquisition to be performed simultaneously. The flow path line-up will be established using the RN Flow Balance. No unique valve line-up or alignments are required beyond the normal scope of the Flow Balance in the combination of these two

procedures. Pump Inservice Test (IWP) data will be taken while the pump is aligned per Flow Balance.

The RN system is not degraded by these tests. The RN pumps are not operated outside the design conditions. The performance of these tests do not make any train of RN inoperable. No unreviewed safety question is created.

PT/0/A/4400/08E, Initial Issue

This procedure allows for the implementation of the Nuclear Service Water (RN) pump portion of the RN Flow Balance without the need to complete the entire RN Flow Balance. In this procedure, only the pump in test is placed in service. The discharge pressure is reduced to the previous full RN Flow Balance discharge pressure, or below, to simulate the conditions of the previous complete RN Flow Balance. The five pump support flows, the Upper Bearing Injection, Lower Bearing Injection, Upper Bearing Oil Cooler, Packing Flush, and Motor Cooler, are all checked against their "Allowable Flow" limits.

If, under this reduced pressure alignment, any of the "Allowable Flow" limits are unachievable by use of the throttle valves, a check against the "Minimum Allowable Flow" limits is performed. If any one of the flows is unachievable, the pump is declared inoperable. If the "Minimum Allowable Flow" limits are met, the restrictive alignment used to achieve the previous RN Flow Balance discharge pressure is removed, and the pump support flows are compared to the "Allowable Flow" limits at the normal operating pressure. Failure to achieve the "Allowable Flow" limits at the reduced pressure, but acceptance at the normal pressure, places the pump in a 30 day operability.

This procedure is determined to be conservative because the pump discharge pressure is required to be equal to or less than that of the previous RN Flow Balance. To do this, options to align flow to the Component Cooling (KC), Diesel Generator Engine Cooling Water (KD), and Containment Spray (NS) Heat Exchangers (Hx) are provided, with the cautions of each addressed. The KC Hx has the possibility of overcooling the Reactor Coolant (NC) pump seals when air is removed from the KC valve actuator, and the valves fail completely open. Thus, the KC Hx is placed in the "Minimum Flow" Mode. The operators are informed that it will be used for pump minimum flow protection, and the other KC Hx of the same unit will be in the "Temperature" Mode. The possibility of KD Hx overcooling exists when the outlet isolation valve is open. This concern is generally limited to winter time lake temperatures and is monitored more frequently at this time. This overcooling is not a concern when the diesels are in operation. All four NS Hxs are included in the flow path options. Flow is not to exceed 4939 gpm for any of the NS Hxs. Caution statements for the KC, KD, and NS Hxs are included in the procedure to alert the test coordinator/operator at the controls of these specific concerns.

The pump flow parameters are addressed in the procedure. Pump run out is a concern if the discharge pressure is reduced significantly. Minimum flow for the pump is also a concern if too few flow paths exist. A caution is included to monitor pump flow such that pump dead heading and pump runnout are not achieved. The variations of KC, KD and NX Hx options provide the flow paths

necessary to achieve the test conditions and remain within the safe flow parameters of the pump.

The RN system is not degraded by this test. The test will not degrade the ability of the system to perform its design function. In the event of an ESF signal, the RN pumps will start and provide all essential flows as described in the FSAR. The fail open KC and KD Hxs are in their Engineered Safety Features ESF positions. The RN pumps are not operated outside of their design conditions. The margin of safety as defined in the bases of the Technical Specifications is not reduced. No unreviewed safety question is created.

PT/0/A/4400/08F, Initial Issue

This procedure allows for the implementation of the Nuclear Service Water (RN) pump portion of the RN Flow Balance without the need to complete the entire RN Flow Balance. In this procedure, only the pump in test is placed in service. The discharge pressure is reduced to the previous full RN Flow Balance discharge pressure, or below, to simulate the conditions of the previous complete RN Flow Balance. The five pump support flows, the Upper Bearing

Injection, Lower Bearing Injection, Upper Bearing Oil Cooler, Packing Flush, and Motor Cooler, are all checked against their "Allowable Flow" limits. If, under this reduced pressure alignment, any of the "Allowable Flow" limits are unachievable by use of the throttle valves, a check against the "Minimum Allowable Flow" limits is performed. If any one of the flows is unachievable, the pump is declared inoperable. If the "Minimum Allowable Flow" limits are met, the restrictive alignment used to achieve the previous RN Flow Balance discharge pressure is removed, and the pump support flows are compared to the "Allowable Flow" limits at the normal operating pressure. Failure to achieve the "Allowable Flow" limits at the reduced pressure, but acceptance at the normal pressure places the pump in a 30 day operability.

This procedure is determined to be conservative because the pump discharge pressure is required to be equal to or less than that of the previous RN Flow Balance. To do this, options to align flow to the Component Cooling (KC), Diesel Generator Engine Cooling Water (KD), and Containment Spray (NS) Heat Exchangers (Hx) are provided with the cautions of each addressed. The KC Hx has the possibility of overcooling the Reactor Coolant (NC) Pump seals when air is removed from the RN/KC Hx inlet valve actuator, causing the valves to fail completely open. Thus, the KC Hxs are placed in the "Minimum Flow" Mode, and the other two KC Hxs are placed in the "Temperature Monitoring" Mode. The operators are informed that the "Mini-Flow" Mode will be used for pump flow support, and the other KC Hxs will be used as the temperature monitoring control.

The possibility of KD Hx overcooling exists when the outlet isolation valve is failed open. This concern is generally limited to winter time lake temperatures and is monitored more frequently at this time. This overcooling is not a concern when the diesels are in operation.

All four NS Hxs are included in the flow path options. Pump run out is a concern if the discharge pressure were to be allowed to drop too low. A

minimum flow limit is stated, and the operators are influential in determining the alignment of the KC, KD, and NS Hx flow paths.

The RN system is not degraded by this test. RN does not initiate any accident conditions in the FSAR. The test will not degrade the ability of the system to perform its design function. In the event of an Engineered Safety Features (ESF) signal, the RN pumps will start and provide all essential flows as described in the FSAR. The fail open position of KC and KD Hxs are in their ESF positions. The RN pumps are not operated outside of their design conditions. The margin of safety as defined in the bases of the Technical Specifications is not reduced in any way. No unreviewed safety question is created.

PT/1/A/4200/09, Change #90

This change adds prerequisites for stroking Refueling Water (FW) valves 1FW27A and 1FW55B to ensure that water is not transferred from the Refueling Water Storage Tank (FWST) to the Reactor Coolant (NC) System. The change reduces the probability of emptying the FWST by gravity feed to either the refueling canal if the reactor vessel head is off, or to the NC system if the head is on. In addition, the Residual Heat Removal (ND) pump on the train being tested is verified to be off prior to the test, so that the ND pump, if aligned to the loops, will not pump the FWST to the NC system. 1FW27A and 1FW55B will be stroked in the same manner as they normally are for the Valve Inservice (IWV) quarterly test, except that the initiating signal will be different. A jumper is placed in the Engineered Safety Features (ESF) test to simulate 1NI185A or 1NI184B open. The margin of safety as defined in the bases to Tech. Specs. will not be reduced. No unreviewed safety question is created.

PT/1/A/4400/03D, Change #2

The Auxiliary Shutdown Panel Supply Units (ASPSU) were designed to be provided cooling water from the Nuclear Service Water system. Nuclear Station Modification (NSM) 11146 will modify the ASPSU's normal cooling medium to be provided from the Component Cooling (KC) System. The ASPSU's will still have the system ability to be provided cooling water during an ASP event from the Nuclear Service Water (RN) System.

The maximum design flow rate supplied to both A and B train ASPSU's from the Component Cooling System at any one time will be no greater than 32 GPM. NSM-11146 will have Catawba Instrument and Electrical group to perform a string check on the applicable KC inlet flow switches to the ASPSU's which regulates KC inlet flow based on refrigerant head pressure. Verification that the addition of this flow to the ASPSU's A and B will not decrease flow to any other components below already established acceptable values will then be confirmed by the successful completion of this procedure.

The capacity of the Component Cooling Pumps far exceeds the additional flow required for the ASPSU's A and B. In the worst case scenario, the KC flow required during a Safety Injection plus Low-Low Refueling Water Storage Tank level is ~6980 gpm. The capacity of one train of KC pumps is ~10,000 gpm.

This procedure can be run during all modes of operation. Section 12.1 (Refueling) will only be run during mode 5. Sections 12.2 and 12.3 can be run in any mode of operation, but only one train at a time will be tested while the opposite train will be supplying the cooling loads required per Technical Specification 3.7.3.

NSM-11146 revised FSAR Sections 9.2.1.2.3 and 9.2.2.4, and FSAR Tables 9.2.2-1 and 9.2.2-1 to reflect the changes associated with supplying cooling water to the ASPSU's A and B with Component Cooling Water. This change does not create an unreviewed safety question.

PT/0/A/4400/08B, Change #1

This restricted procedure change is to provide a method of adding the recently cleaned Component Cooling (KC) 1B Heat Exchanger (Hx) to the Nuclear Service Water (RN) flow alignment without performi[illegible] flow balance on the entire B train.

Presently, Containment Spray (NS) 1B and KC 1B Hxs are tagged out and not in the RN system alignment. KC 1B is needed to be functional for drain down following refueling. Due to the fact NS 1B is tagged out of service and will remain so for the next one to two weeks when KC 1B is required, a full flow balance alignment is not available. To ensure Unit 2's operability is not in question when KC 1B is valved in following cleaning, a design analysis was performed for existing plant conditions (Unit 2 in Mode 1 and Unit 1 in Mode 5 or 6, approximate day 45 in 1EOC5) to determine an acceptable reduced flow for KC 1B that would not jeopardize the operability of Unit 2 B Train. The error adjusted flow (by Design Analysis) through KC 1B was determined to be 3150 gpm. The difference between the Test Acceptance Criteria (TAC) Sheet operable flow requirement of 5280 gpm and 3150 gpm is 2130 gpm. This exceeds the required flow of 571 gpm that would have been established to NS 1B Hx by RN 2B Pump, required to simulate Auxiliary Feedwater (CA) and Containment Penetration Valve Injection Water System (NW) make up to Unit 2. Additionally, by verifying acceptable flow, and at the same time not adjusting the throttle valves to KC 2B, NS 2B, Diesel Generator Jacket Water Cooling (KD) 2B and Control Area Ventilation/Chilled Water (VC/YC) B Train, this 3150 gpm to KC 1B is established at the reduced pump discharge pressure. Therefore, it is determined that the adjusted flow paths are conservative for the present plant condition and 2B RN is demonstrated operable.

The RN system is not degraded by these tests. The RN pumps are not operated outside the design conditions. No equipment malfunctions are created. The performance of these tests do not make any train of RN inoperable. No unreviewed safety question is created.

MP/0/A/7150/29 Retype, Changes 0 to 3 Incorporated

Section 11.0 of this procedure has been completely rewritten to include lessons learned and improved techniques for working on the Nuclear Service Water (RN) pump. The pumps are being modified to remove the lube injection

system on the pumps. The procedure shows two methods for repacking the pump. As soon as all pumps have been modified, the old method will be deleted from the procedure. In step 11.10.13, the option is now being given to use the Optalign method for aligning the pump and motor.

Tech Spec. 3/4.7.4 is affected by this procedure. Operations has the responsibility and the procedures for compliance with these Tech. Specs. Maintenance will be performed on this pump when Tech. Specs. allow, per Operation's procedures. This rewrite will clarify and assure that maintenance activities will return the pumps to as-designed conditions.

The changes made by this rewrite have been reviewed against approved vendor manuals, design documents, and station procedures to ensure that the corrective maintenance controlled by this procedure will return the pump to as-built/as-designed condition. These actions will ensure the pump's compliance with FSAR accident analysis. Therefore, no unreviewed safety question exists.

PT/0/A/4400/08A, Change #2

The purpose of this procedure change is to allow for the Nuclear Service Water (RN) Flow Balance: Train A, the Nuclear Service Water Pump Train A Performance Test, and the RN Pump 1A flow versus pressure data acquisition to be performed simultaneously. The flow path line-up will be established using the RN Flow Balance. No unique valve line-up or alignments are required beyond the normal scope of the Flow Balance in the combination of these two procedures.

Before the completion of the RN Flow Balance on Pump 1A, a flow versus pressure data profile will be recorded. This is new enclosure 13.8 of the Flow Balance. While in the RN Flow Balance, the ordinary valve line-up is such that the flow conditions are near the maximum flow paths available to RN in the one pump/isolated train alignment. Thus, by opening the final Containment Spray (NS) Heat Exchanger (Hx), a flow path of 2 NS, 2 Component Cooling (KC), 2 Diesel Generator Engine Jacket Cooling Water (KD) Hxs, 1 Control Area Ventilation/Chilled Water (VC/YC) chiller and various other flows maximize the flow for that data point of the pump curve profile. Following the maximum flow alignment, the return to normal operating alignment will be performed. By gradually closing off each of the flow paths, a flow versus pressure profile may be constructed. This activity is the normal return to the "As Found" condition. The flow balance safety positions will not be affected in any way. The profile will be obtained as the system is returned to the normal operating alignment for Train A. Cautions to avoid the pump runout condition of 26,000 gpm and mini-flow dead heading of 8,500 gpm are emphasized throughout the procedure steps.

One concern beyond the scope of the RN Flow Balance is the possibility of a second pump start while the flow paths have been reduced in the gathering of low flow, high pressure data. On the possibility of a low pit level, a signal for the second train pump to start could exceed the immediate mini-flow alignment. The operators at the controls have the capability to open the NS Hxs from the control board. The Test Coordinator has the capability of

removing air from 1RN291 and 2RN291 immediately allowing full flow to the KC Hxs. The KC Hxs would then be returned to their flow balance, fail safe positions. These cautions are also highlighted within the procedure. If a failure of the instrument air (VI) system occurred, the KC Hxs would be aligned to their fail safe position, removing any concerns of inadequate mini-flow protection for the two pumps. The KC Hxs can adequately provide a flow path to ensure no dead heading occurs for both pumps in operation.

The RN system is not degraded by these tests. The RN pumps are not operated outside the design conditions. The performance of these tests do not make any train of RN inoperable. No unreviewed safety question is created.

MP/0/A/7150/94, Initial Issue

Cable - type tube dampers are installed in steam generator tubes to limit the amount of their vibration, and to limit their range of motion should the tube completely sever. Tubes which have dampers installed are also removed from service by plugging both the hot leg and cold leg ends.

The installation of this damper will not degrade a primary system pressure boundary. Tube rupture is already analyzed in the FSAR. The presence of the damper will limit tube movement should it completely sever. No operating limits or assumptions made in any safety analyses have been violated.

There are no unreviewed safety questions concerning the installation of cable - type tube dampers into the steam generators in accordance with the above referenced procedure.

TN/1/A/1242/00/09A, Initial Issue

Nuclear Station Modification (NSM) CN-11242, Rev. 0, will modify the control circuit wiring on valves 1CA002, 004, 006, 1FW001A, 032B, 1KC003A, 018B, 228B, 230A, 305B, 315B, 320A, 332B, 333A, 338B, 424B, 425A, 1ND032A, 065B, 1NF233B, 1RF389B, 447B, and 457B to provide "limited actuated" torque switch bypass contacts which can be adjusted independently of indications or interlocks and provide data to complete Motor Operated Valve (MOV) testing of the valves. The MOV testing information included in the NSM will supersede the old torque switch setting values and replace them with thrust values. The settings are selected, set, and maintained correctly to accommodate the maximum differential pressure expected on the valve during both normal and abnormal events within the design basis. The new thrust values ensure the valve will operate during normal and abnormal events by setting limitations on Total Thrust, ΔP Thrust, and Packing Load.

This procedure will control work being performed on Fire Protection (RF) valve 1RF389B. Instrument and Electrical (IAE) will perform all work at the valve. IAE will rewire the rotors, set up the switch rotors, verify add-on-pak switch setup, and perform MOV testing of the valve.

TN/1/A/1242/00/09A requires power to be removed from 1RF389B during implementation. 1RF389B is the Containment Building Hose Rack Isolation Valve. This valve may be operated manually at the valve or from the Control Room. This valve is a containment isolation valve described in Tech. Spec. 3.6.3. Per this Tech. Spec., this valve is not required to be operable for the containment isolation function in Modes 5, 6, and No Mode. This procedure will be performed with the unit in these modes.

This valve is kept normally closed, to keep the hose rack header dry. Thus, removing power from the valve will render the containment hose rack(s) inoperable. In addition, in order to perform MOV and type "C" leak rate testing on the valve, 1RF-254 and 1RF-923 are required to be closed. This disables the following equipment:

1. Sprinklers in the Unit 2 Auxiliary Building Cable Room Corridor, Unit 2 Auxiliary Building Battery Room Corridor, and the Unit 2 Auxiliary Feedwater Pump Room.

2. Hose stations in Unit 1 lower containment.

3. Hose stations in the Auxiliary Building elevations 543', 560', 577', and 594'.

By isolating 1RF-389B, the systems are incapable of automatic action (i.e., sprinkler actuation.) Fire Protection operability is addressed in the Selected Licensee Commitment Manual. Fire protection equipment is generally required to be operable whenever equipment in the area is required to be operable. To compensate for 1RF-254 and 1RF-923 being closed, a Compensatory Action statement was written. An operator with a hand-held radio was to be stationed at valves 1RF-254 and 1RF-923. Upon notification from the Fire Protection Console Operator, valves 1RF-254 and 1RF-923 would be opened.

Prior to returning the valve to service, a functional verification and retest will be performed to verify valve operability. No unreviewed safety questions are created by implementation of this procedure.

TN/1/A/1242/00/10A, Initial Issue

Nuclear Station Modification (NSM) CN-11242, Rev. 0, will modify the control circuit wiring on valves 1CA002, 004, 006, 1FW001A, 032B, 1KC003A, 018B, 228B, 230A, 305B, 315B, 320A, 332B, 333A, 338B, 424B, 425A, 1ND032A, 065B, 1NF233B, 1RF389B, 447B, and 457B to provide "limited actuated" torque switch bypass contacts which can be adjusted independently of indications or interlocks and provide data to complete Motor Operated Valve (MOV) testing of the valves. The MOV testing information included in the NSM will supersede the old torque switch setting values and replace them with thrust values. The settings are selected, set, and maintained correctly to accommodate the maximum differential pressure expected on the valve during both normal and abnormal events within the design basis. The new thrust values ensure the valve will operate during normal and abnormal events by setting limitations on Total Thrust, ΔP Thrust, and Packing Load.

This procedure will control work being performed on Fire Protection (RF) valve 1RF447B. Instrument and Electrical (IAE) will perform all work at the valve. IAE will rewire the rotors, set up the switch rotors, verify add-on-pak switch setup, and perform MOV testing of the valve.

TN/1/A/1242/00/10A requires power to be removed from 1RF447B during implementation. 1RF447B is the Containment Building Sprinkler Isolation Valve. This valve may be operated manually at the valve or from the Control Room. This valve is a containment isolation valve described in Tech. Spec. 3.6.3. Per this Tech. Spec., this valve is not required to be operable for the containment isolation function in Modes 5, 6, and No Mode.

This valve is kept normally closed, to keep the sprinkler header dry. Thus, removing power from the valve will render the containment sprinkler system inoperable. Selected Licensee Commitment 16.9-2 requires that the sprinkler system be operable whenever equipment protected by the sprinkler system is required to be operable. This TN was implemented during No Mode of the Unit 1 refueling outage. Thus, no equipment protected by this sprinkler system was required to be operable while the work was being performed, and the containment isolation function was not required.

Prior to returning the valve to service, functional verification and retest were performed to verify valve operability. An unreviewed safety question does not exist.

TN/1/A/1242/00/11A, Initial Issue

Nuclear Station Modification (NSM) CN-11242, Rev. 0, will modify the control circuit wiring on valves 1CA002, 004, 006, 1FW001A, 032B, 1KC003A, 018B, 228B, 230A, 305B, 315B, 320A, 332B, 333A, 338B, 424B, 425A, 1ND032A, 065B, 1NF233B, 1RF389B, 447B, and 457B to provide "limited actuated" torque switch bypass contacts which can be adjusted independently of indications or interlocks and provide data to complete Motor Operated Valve (MOV) testing of the valve. The MOV testing information included in the NSM will supersede the old torque switch setting values and replace them with thrust values. The settings are selected, set, and maintained correctly to accommodate the maximum differential pressure expected on the valve during both normal and abnormal events within the design basis. The new thrust values ensure the valve will operate during normal and abnormal events by setting limitations on Total Thrust, ΔP Thrust, and Packing Load.

This procedure will control work being performed on Fire Protection (RF) valve 1RF457B. Instrument and Electrical (IAE) will perform all work at the valve. IAE will rewire the rotors, set up the switch rotors, verify add-on-pak switch setup, and perform MOV testing of the valve.

TN/1/A/1242/00/11A requires power to be removed from 1RF457B during implementation. 1RF457B is the Annulus Sprinkler Header Isolation Valve. This valve may be operated manually at the valve or from the Control Room. Removing power from the valve will cause it to fail as is. This may render the annulus sprinkler system inoperable. Selected Licensee Commitment 16.9-2 requires that the sprinkler system be operable whenever equipment protected by the sprinkler system is required to be operable. This TN was implemented

during No Mode of the Unit 1 refueling outage. Thus, no equipment protected by this sprinkler system was required to be operable while the work was being performed.

Prior to returning the valve to service, a functional verification and retest were performed to verify valve operability. No unreviewed safety question was created by implementation of this procedure.

TT/1/A/9200/062, Initial Issue

This procedure will initiate four transient tests to verify proper operation of the modifications performed on Feedwater and Turbine Control.

The first transient test will simulate the loss of one of the two breakers that feeds the Digital Turbine Control System. The system was designed so that two breakers feed power to the system. In the event that one of those breakers opens, the remaining breaker will be able to feed the system without loss of turbine control or power to the Operator's control panel. The tested power system is considered to be a Non-1E Class system by the FSAR. This turbine system is not required for safe shutdown of the reactor. The breakers that are opened are not safety related and only affect the Digital Turbine Control. The system shall be tested per its design and will not be placed under abnormal conditions. No other system will be affected.

The transient tests described are considered to be Condition I transients. Condition I transients are those that are considered normal and expected transients during power operation. Also, Condition I transients cannot be expected to cause Conditions II, III, or IV, but do affect the consequences of those fault conditions.

The second transient test initiates a 10% step decrease in load by dropping 120 MWe from the turbine. The third transient test initiates a loss of a feedpump at 100% power to verify Turbine Control will run the system back to 65% Load, and Digital Feedwater will control Steam Generator (S/G) levels within alarm setpoints. The fourth transient test simulates a loss of electrical load at a power level below P-9 to verify that the Turbine Control will reduce load without tripping the turbine, and the Feedwater control shall maintain the S/G levels. This transient test is performed by disconnecting the system from the grid by opening the main generator breakers. The proceeding evaluation will apply for all three transients since the same systems are affected and called upon during the transient tests.

Certain conditions must be met prior to performing any of the three transient tests. First, the plant shall be in normal operating condition for safe plant operation. The plant shall also meet stability criteria as specified in the procedure prior to beginning the tests. All three transient tests were required for initial Unit startup and are described by the FSAR under Chapter 14, Section 14.5. The procedure will not test any system beyond its design capabilities. No configuration of the plant will be made that could jeopardize equipment operation. All trips and protection systems required by Tech. Specs. and Chapter 15 of the FSAR shall be operable prior to performing the test. The affected systems are considered to be non-safety related and

are not required for the safe shutdown of the reactor. The system is being operated as designed.

Protection trip functions for the turbine, such as Overspeed Protection, are not bypassed by the test. The reactor protection trip functions shall also be maintained during the test to ensure safe shutdown to the reactor. The tested systems are not configured in any way to challenge the systems beyond their design.

Since plant configuration will be maintained in a normal alignment, no Tech. Spec. action statements shall be entered. The only possible affected Tech. Spec. occurs during load rejection, and the Rod Insertion Limit Annunciator is picked up. The procedure recognizes this and provides for corrective action. Per the Tailgate package, the operators are notified beforehand that they will need to take necessary actions per OP/1/A/6150/09 to clear this alarm.

Based upon this evaluation, no Unreviewed Safety Question exists for this procedure.

TT/1/A/4150/01A, Initial Issue

This procedure will be used to connect a Westinghouse digital reactivity computer (DRC) to the Boron Dilution Mitigation System (BDMS) (ENC) train A raw detector signal output. This will allow an alternate method for connecting the DRC which avoids taking a Power Range (P/R) detector out of service. The procedure has an option in it to either connect the DRC locally at the ENC amplifier box, or to allow the DRC to be remotely connected via the Operator Aid Computer (OAC) Wide Range (W/R) flux signal cable.

The BDMS is required in Modes 3 to 5. This test will not render it inoperable while in these modes. ENC train A will become technically inoperable when the connections are made between the QA-1 ENC circuit and the non-QA DRC (local connection) or jumper to the OAC cable (remote connection). However, this will [illegible] be done in Mode 2 following the train being defeated. The single failur[illegible] [illegible]iteria for the ENC system is still satisfied because train separat[illegible] will still be maintained. If the Unit enters Mode 3 with train A inoperab[illegible], the procedure has steps to restore it to operability. The Tech. Specs. allow one train to be inoperable for 48 hours in Modes 3 to 5. Both trains may be inoperable in Modes 3 to 5 for 12 hours. Restoring the train can be done in less than six hours.

While in Mode 2, no FSAR accident takes credit for the ENC system. Train A will be rendered inoperable only in the technical sense. It still will be functional. Since it is blocked in Mode 2, no equipment important to safety is affected. The margin of safety in the Tech. Specs. is not reduced. Adequate precautions are in place to restore the train within the Tech. Spec. allowable time if the Unit enters Mode 3. No unreviewed safety question is created.

TT/1/A/4150/01B, Initial Issue

This procedure will be used to connect a Westinghouse digital reactivity computer (DRC) to the Boron Dilution Mitigation System (BDMS) (ENC) train B raw detector signal output. This will allow an alternate method for connecting the DRC which avoids taking a Power Range (P/R) detector out of service. The procedure has an option in it to either connect the DRC locally at the ENC amplifier box, or to allow the DRC to be remotely connected via the Operator Aid Computer (OAC) Wide Range (W/R) flux signal cable.

The BDMS is required in Modes 3 to 5. This test will not render it inoperable while in these modes. ENC train B will become technically inoperable when the connections are made between the QA-1 ENC circuit and the non-QA DRC (local connection) or jumper to the OAC cable (remote connection). However, this will only be done in Mode 2 following the train being defeated. Train B only provides W/R flux indication to the Control Room. Train A W/R indication will still be available. The single failure criteria for the ENC system is still satisfied because train separation will still be maintained. If the Unit enters Mode 3 with train B inoperable, the procedure has steps to restore it to operability. The Tech. Specs. allow one train to be inoperable for 48 hours in Modes 3 to 5. Restoring the train can be done in less than six hours.

While in Mode 2, no FSAR accident takes credit for the ENC system. Train B will be rendered inoperable only in the technical sense. It still will be functional. Since it is blocked in Mode 2, no equipment important to safety is affected. The margin of safety in the Tech. Specs. is not reduced. Adequate precautions are in place to restore the train within the Tech. Spec. allowed time if the Unit enters Mode 3. No unreviewed safety question is created.

TN/1/A/1229/00/01A, Initial Issue

Nuclear Station Modification (NSM) CN-11229, Rev. 0 replaces the Unit 1 Post Accident Liquid Sample System (PALSS) with the PALSS II+ system which will be more reliable and require less maintenance. The new system has a simple layout, fewer components and a less complicated control panel. This procedure controls the installation of Nuclear Sampling (NM) valves 1NM801, 1NM805, and 1NM806 and associated piping. These valves will be used to provide system isolation to support installation of the PALSS II+ system.

In order to perform this procedure, the Unit 1 Reactor Coolant (NC) A and C Hot Leg sample lines and Residual Heat Removal (ND) Pump Discharge sample lines must be isolated. Isolation of these lines impacts NC system sampling requirements for Tech. Specs. 3/4.4.7 and 3/4.4.8. Tech. Specs. require that the NC system be sampled and analyzed at least once per 72 hours to verify system chemistry and specific activity limits. Implementation of this procedure will be coordinated with Catawba Chemistry sampling intervals to ensure compliance with Tech. Specs.

During implementation of this procedure, the sample input tubing to the PALSS panel will be disconnected and modified to allow installation of the isolation valves and associated equipment. This tubing will remain disconnected from

the PALSS panel after this procedure is completed and will be reconnected to the PALSS II+ panel installed under TN/1/A/1229/00/02A. While the input tubing is disconnected, no sampling capabilities are available through the PALSS panel. The PALSS panel allows sampling of the NC and ND systems and containment sump during accident conditions for laboratory analysis. With the PALSS panel out of service, the NC and ND systems and containment sump could be manually sampled through the NM system piping and Radiation Monitors (EMFs) to provide indication of system and containment conditions during accident situations. Therefore, in the event of an accident during implementation of this procedure, sampling capabilities are available. Also, other means, such as station EMFs and High Range Containment Monitors, are available for evaluating core damage and containment environment during accident conditions. No other systems or equipment important to plant safety are affected by these isolations.

After installation of the isolation valves and associated equipment, the modified tubing and piping will be visually inspected at system temperature and pressure to verify piping integrity. This will ensure NM system integrity to support subsequent sampling activities.

Sampling capabilities are available during implementation of this procedure, and no other equipment required to support safe operation of the station is affected. Based on the above, no unreviewed safety questions are created by this procedure.

TN/1/A/1229/00/02A, Initial Issue

Nuclear Station Modification (NSM) CN-11229, Rev. 0 replaces the Unit 1 Post Accident Liquid Sample System (PALSS) with the PALSS II+ system which will be more reliable and require less maintenance. The new system has a simpler layout, fewer components and a less complicated control panel. This procedure controls the installation of the new PALSS II+ system, installs the associated piping and support equipment, and completes all associated electrical activities.

In order to perform this procedure, the Unit 1 PALSS panel and its associated control panel must be removed from service and all inputs to the panel isolated. This includes electrical isolation of the PALSS panel and sump pump and mechanical isolation of Component Cooling (KC), Instrument Air (VI), Demineralized Water (YM), and the PALSS Nitrogen supply. These isolations affect inputs to the PALSS and Post Accident Containment Sample System (PACSS) and associated components only, and will not affect the operation of any other equipment important to plant safety. The Auxiliary Building Ventilation (VA) duct to the PALSS panel will be disconnected with the VA system in service. This has no impact on VA system operability.

The PALSS and PACSS are part of the Nuclear Sampling (NM) system which provides various system sampling capabilities. The PALSS and PACSS allow sampling of the Reactor Coolant (NC) and Decay Heat Removal (ND) systems, the containment sump, and the containment atmosphere during accident conditions for laboratory analysis. With the PALSS and PACSS out of service, the NC and ND systems and containment atmosphere could be manually sampled through the NM system piping and Radiation Monitors (EMFs) to provide indication of system

and containment conditions during accident situations. Therefore, in the event of an accident during implementation of this procedure, sampling capabilities are available. Also, other means, such as station EMFs and High Range Containment Monitors, are available for evaluating core damage and containment environment during accident conditions.

After installation, the associated PALSS II+ tubing and piping will be visually inspected at system temperature and pressure to verify piping integrity. Also, the PALSS II+ system will be tested in accordance with the Post Accident Liquid Sampling System Periodic Test to verify system performance and operability. Continuity checks will be performed on all new and revised circuits to ensure operability. After completion of this procedure, the PALSS II+ system pressure boundary will be restored and the PALSS II+ system and PACSS returned to service.

System isolations are not required, and controls are provided to ensure compliance with Technical Specifications. Post Modification Testing is not required for this modification. Based on the above, no unreviewed safety questions are created by this procedure.

TN/1/A/1005/02/01A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the removal of snubbers deleted from Math Model KCX. This math model includes supports on the Component Cooling (KC) system. These supports will either be deleted from the system or revised to a different configuration.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model KCX has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined this work may be done while the affected system(s) are operable, provided that all the support modifications for the entire math model are completed within the 72 hours allowed by the Technical Specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified under any combination of snubbers removed, provided the modifications to the existing supports have been completed. This procedure is written to complete all modifications before deleting any supports. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of modified supports provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure

will provide the necessary controls to ensure compliance with all Technical Specification requirements.

System isolations are not required, and controls are provided to ensure compliance with Technical Specifications. Post Modification Testing is not required for this modification. Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/02A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the removal of snubbers deleted from Math Model KCF. This math model includes supports in the Component Cooling (KC) system. These supports will either be deleted from the system or revised to a different configuration.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model KCF has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the Technical Specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified under any combination of snubbers removed, provided the modifications to the existing supports have been completed. This procedure is written to complete all modifications before deleting any supports. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of modified supports provided the affected piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all Technical Specification requirements.

System isolations are not required, and controls are provided to ensure compliance with Technical Specifications. Post Modification Testing is not required for this modification.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/05A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the rework of supports on Math Model KCA. This math model includes supports only on the Component Cooling (KC) system.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model KCA has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined this work may be done while the affected system(s) are operable provided all the support modifications for the entire math model are completed within the 72 hours allowed by the Technical Specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified, provided the modifications to the existing supports have been completed. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of modified supports provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

System isolations are not required, and controls are provided to ensure compliance with Technical Specifications. Post Modification Testing is not required for this modification.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/07A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the rework of supports on Math Model KCB. This math model includes supports only on the Component Cooling (KC) system.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model KCB has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require

numerous analyses. For this reason, Design has determined that this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the Technical Specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified, provided the modifications to the existing supports have been completed. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of modified supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all Technical Specification requirements.

System isolations are not required, and controls are provided to ensure compliance with Technical Specifications. Post Modification Testing is not required for this modification.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/08A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the rework of supports on Math Model KCI. This math model includes supports only on the Component Cooling (KC) system.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model KCI has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined that this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the Technical Specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified, provided the modifications to the existing supports have been completed. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of modified supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all Technical Specification requirements.

System isolations are not required. Controls are provided to ensure compliance with Technical Specifications. Post Modification Testing is not required for this modification.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/09A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the rework of supports on Math Model NDA. This math model includes supports only on the Residual Heat Removal (ND) system.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model NDA has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the Technical Specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified, provided the modifications to the existing supports have been completed. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of modified supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all Technical Specification requirements.

System isolations are not required. Controls are provided to ensure compliance with Technical Specifications. Post Modification Testing is not required for this modification.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/11A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the rework of supports on Math Model NDN.

This math model includes supports only on the Residual Heat Removal (ND) system.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model NDN has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined that this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the Technical Specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified, provided the modifications to the existing supports have been completed. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of modified supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all Technical Specification requirements.

System isolations are not required. Controls are provided to ensure compliance with Technical Specifications. Post Modification Testing is not required for this modification.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/12A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the rework of supports in various systems. This support analysis includes supports only on the Hydrogen Bulk Storage (GS), Component Cooling (KC), Boron Recycle (NB), Reactor Coolant (NC), Chemical and Volume Control (NV), and Solid Waste Control (WS) systems.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. This Math Model has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed, because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined this work may be done while the affected system(s) are operable, provided that all the support

modifications for the entire math model are completed within the 72 hours allowed by the Technical Specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified, provided the modifications to the existing supports have been completed. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated that they could justify operable status of any configuration of modified supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all Technical Specification requirements.

System isolations are not required. Controls are provided to ensure compliance with Technical Specifications. Post Modification Testing is not required for this modification.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/16A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the rework of supports on Math Model NBD. This math model includes supports only on the Boron Recycle (NB), Decay Heat Removal (ND), and Safety Injection (NI) systems.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. This Math Model has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the Technical Specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified, provided the modifications to the existing supports have been completed. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated they could justify operable status of any configuration of modified supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all Technical Specification requirements.

System isolations are not required. Controls are provided to ensure compliance with Technical Specifications. Post Modification Testing is not required for this modification.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/18A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the rework of supports on Math Model YCA. This math model includes supports only on the Chilled Water (YC) system.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. This Math Model has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the Technical Specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified, provided the modifications to the existing supports have been completed. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated that they could justify operable status of any configuration of modified supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all Technical Specification requirements.

System isolations are not required. Controls are provided to ensure compliance with Technical Specifications. Post Modification Testing is not required for this modification.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/20A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the rework of supports on Math Model RNE. This math model includes supports only on the Nuclear Service Water (RN) system.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. This Math Model has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined that this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the Technical Specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified, provided the modifications to the existing supports have been completed. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated that they could justify operable status of any configuration of modified supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all Technical Specification requirements.

System isolations are not required. Controls are provided to ensure compliance with Technical Specifications. Post Modification Testing is not required for this modification.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/21A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the rework of supports on Math Model BWD. This math includes supports only on the Steam Generator Blowdown (BB), Steam Generator Wet Lay-up (BW), and Make-up Demineralized Water (YM) systems.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. This Math Model has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined that this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the Technical Specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping

could be qualified, provided the modifications to the existing supports have been completed. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated that they could justify operable status of any configuration of modified supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all Technical Specification requirements.

System isolations are not required. Controls are provided to ensure compliance with Technical Specifications. Post Modification Testing is not required for this modification.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/23A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the rework of supports on Math Model CAF. This math model includes supports only on the Auxiliary Feedwater (CA) and Main Feedwater (CF) systems.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. This Math Model has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined that this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the Technical Specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified, provided the modifications to the existing supports have been completed. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated that they could justify operable status of any configuration of modified supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all Technical Specification requirements.

System isolations are not required. Controls are provided to ensure compliance with Technical Specifications. Post Modification Testing is not required for this modification.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/1005/02/29A, Initial Issue

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the rework of supports on Math Model BWC. This math model includes supports only on the Steam Generator Blowdown (BB), Steam Generator Wet Layup (BW) and Makeup Demineralized Water (YM) systems.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. This Math Model has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined that this work may be done while the affected system(s) are operable, provided all the support modifications for the entire math model are completed within the 72 hours allowed by the Technical Specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified, provided the modifications to the existing supports have been completed. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated that they could justify operable status of any configuration of modified supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all Technical Specification requirements.

System isolations are not required. Controls are provided to ensure compliance with Technical Specifications. Post Modification Testing is not required for this modification.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/2/A/0632/00/01A, Initial Issue

Nuclear Station Modification (NSM) CN-20632, Rev. 0, will revise the controls on the VF (Fuel Building Ventilation) system so that when a trip 2 signal is received from Radiation Monitors (EMF's) 35, 36, 37, and 42, the system will go into filter mode instead of shutting down the fans. This procedure will provide guidelines for isolation, electrical work, restorations, and functional verification of the modification.

This procedure may be implemented with Unit Two in any mode. The isolations that will have to be performed to implement this procedure will affect the VF System (Fuel Building Ventilation), VE System (Annulus Ventilation), and two of the Reactor Vessel Head Vent valves.

During the implementation of this procedure, as a result of the isolations, both trains of VF will be inoperable. Per Tech. Spec. 3.9.11, this is allowed as long as no operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool is taking place. No fuel movement will be taking place during the implementation of this procedure.

Also as a result of the isolations, VE Train A will be inoperable due to power being removed from VE Fan 2A, damper 2AVS-D-4 (2VE-4, VE Fan 2A Miniflow Isolation), and 2AVS-D-5 (VE Fan 2A Isolation Damper). Per Tech. Spec. 3.6.1.8, one train of VE may be inoperable for 7 days while the Unit is in Modes 1, 2, 3, or 4.

The isolations for this procedure effect Tornado Isolation Dampers 2FPX-D-3A, 2FPX-D-6A, 2FPX-D-3B, 2FPX-D-6B, and 2AVS-D-5. As a result of the isolations, Fuel Pool Exhaust dampers 2FPX-D-3A, 2FPX-D-6A, 2FPX-D-3B, and 2FPX-D-6B will fail open. However, these dampers have a back-up air supply tank that would close these dampers if the Tornado Isolation pushbutton was depressed on Control Board 2MC5. Damper 2AVS-D-5 is also a fail open damper, so this damper will be gagged closed before the isolations are made to protect the plant in case of a tornado. VE Train A will be inoperable during the implementation of this procedure, so having 2AVS-D-5 closed will not be an operational concern.

The isolations made for Control Board plug P-04-16 also affect Reactor Head Vent valves 2NC251B and 2NC252B. Two of the four Reactor Head Vent valves, 2NC252B and 2NC253A (in se s), are closed with power removed during normal operation. Therefore, a single failure would affect only one of the two powered valves, 2NC250A and 2NC251B. Valves 2NC251B and 2NC252B will have power removed from them to implement this procedure. 2NC251B will be tagged closed for this procedure, making one of the reactor coolant vent paths inoperable. To comply with Tech. Spec. 3.4.11, all valves in this vent path, 1NC250A, 1NC251B, 1NC252B, and 1NC253A, will be tagged closed. Per Tech. Spec. 3.4.11. ne reactor coolant vent path may be inoperable for 30 days while the Unit is in Modes 1, 2, 3, or 4.

All modified circuits will be functionally tested prior to returning them to service. No unreviewed safety questions are created.

TN/2/A/0611/00/01A, Initial Issue

Nuclear Station Modification (NSM) CN-20611, Rev. 0 replaces the Unit 2 Post Accide-t Liquid Sample System (PALSS) with the PALSS II+ system which will be more reliable and require less maintenance. The new system has a simpler layout, fewer components and a less complicated control panel. This procedure controls the installation of Nuclear Sampling (NM) System valves 2NM801, 2NM805, and 2NM806 and associated piping. These valves will be used to provide system isolation to support installation of the PALSS II+ system.

In order to perform this procedure, the Unit 2 Reactor Coolant (NC) A and C Hot Leg sample lines and Residual Heat Removal (ND) Pump Discharge sample lines must be isolated. Isolation of these lines impacts NC system sampling requirements for Tech. Specs. 3/4.4.7 and 3/4.4.8. Tech. Specs. require that the NC system be sampled and analyzed at least once per 72 hrs. to verify system chemistry and specific activity limits. Implementation of this procedure will be coordinated with Catawba Chemistry sampling intervals to ensure compliance with Tech. Specs.

During implementation of this procedure, the sample input tubing to the PALSS panel will be disconnected and modified to allow installation of the isolation valves and associated equipment. This tubing will remain disconnected from the PALSS panel after this procedure is completed and will be reconnected to the PALSS II+ panel installed under TN/2/A/0611/00/02A. While the input tubing is disconnected, no sampling capabilities are available through the PALSS panel. The PALSS panel allows sampling of the NC and ND systems and containment sump during accident conditions for laboratory analysis. With the PALSS panel out of service, the NC and ND systems and containment sump could be manually sampled through the NM system piping and Radiation Monitors (EMFs) to provide indication of system and containment conditions during accident situations. Therefore, in the event of an accident during implementation of this procedure, sampling capabilities are available. Also, other means, such as station EMFs and High Range Containment Monitors, are available for evaluating core damage and containment environment during accident conditions. No other systems or equipment important to plant safety are affected by these isolations.

After installation of the isolation valves and associated equipment, the modified tubing and piping will be visually inspected at system temperature and pressure to verify piping integrity. This will ensure NM system integrity to support subsequent sampling activities.

No new failure modes or operating characteristics are introduced by this procedure. Sampling capabilities are available during implementation of this procedure. No other equipment required to support safe operation of the station is affected. No unreviewed safety questions are judged to be involved or created by this procedure.

TN/2/A/0611/00/02A, Initial Issue

Nuclear Station Modification (NSM) CN-20611, Rev. 0 replaces the Unit 2 Post Accident Liquid Sample System (PALSS) with the PALSS II+ system which will be more reliable and require less maintenance. The new system has a simpler layout, fewer components and a less complicated control panel. This procedure controls the installation of the new PALSS II+ system, installs the associated piping and support equipment, and completes all associated electrical activities.

In order to perform this procedure, the Unit 2 PALSS panel and its associated control panel must be removed from service and all inputs to the panel isolated. This includes electrical isolation of the PALSS panel and sump pump, and mechanical isolation of Component Cooling (KC), Instrument Air (VI), Demineralized Water (YM), and the PALSS Nitrogen supply. These isolations

affect inputs in the PALSS and Post Accident Containment Sample System (PACSS) and associated components only, and will not affect the operation of any other equipment important to plant safety. The Auxiliary Building Ventilation (VA) duct to the PALSS panel will be disconnected with the VA system in service. This has no impact on VA system operability.

The PALSS and PACSS are part of the Nuclear Sampling (NM) system which provides various system sampling capabilities. The PALSS and PACSS allow sampling of the Reactor Coolant (NC) and Decay Heat Removal (ND) systems, the containment sump, and containment atmosphere during accident conditions for laboratory analysis. With the PALSS and PACSS out of service, the NC and ND systems and containment atmosphere could be manually sampled through the NM system piping and Radiation Monitors (EMFs) to provide indication of system and containment conditions during accident situations. Therefore, in the event of an accident during implementation of this procedure, sampling capabilities are available. Also, other means, such as station EMFs and High Range Containment Monitors, are available for evaluating core damage and containment environment during accident conditions.

After installation, the associated PALSS II+ tubing and piping will be visually inspected at system temperature and pressure to verify piping integrity. Also, the PALSS II+ system will be tested in accordance with the Post Accident Liquid Sampling System Periodic Test to verify system performance and reliability. Continuity checks will be performed on all new and revised circuits to ensure operability. After completion of this procedure, the PALSS II+ system pressure boundary will be restored and the PALSS II+ system and PACSS returned to service.

No new failure modes or operating characteristics are introduced by this procedure. Sampling capabilities are available during implementation of this procedure. No other equipment required to support safe operation of the station is affected. No unreviewed safety questions are judged to be involved or created by this procedure.

PT/0/A/4600/06B, Initial Issue

This procedure supersedes unit specific procedures PT/1/A/4600/06B and PT/2/A/4600/06B. It is identical to these procedures with the exception of the following changes:

1. Step 1.2, 11.1, and 12.15 - The reference to the Data Book was revised to make it applicable to either unit.

2. Step 6.4 - A new step was added to provide warning of Radiation Monitor (EMF) alarms.

3. Step 8.3 - The step was revised to ensure that the applicable unit is identified to Radiation Protection (R.P., previously "Health Physics") for posting.

4. Step 8.4 - The step was changed to specify use of Unit Containment Entry Log.

5. Step 12.6 - The step was revised to allow Incore Detector System (ENA) detectors to be returned to Storage and R.P. posting to be removed if new detector setpoints are required but further ENA operation is not presently desired.

6. Step 12.10 - New steps were added to provide for re-posting of appropriate areas prior to subsequent ENA operation.

7. Step 12.14 - Notification of Operations as well as Radiation Protection was specified.

The ENA System is operated at all times in accordance with OP/0/A/6150/07, "Incore Instrumentation". Additional measures are adopted to ensure that no personnel radiation exposure results from ENA detector operation. No unreviewed safety questions are created by this procedure.

PT/0/A/4600/06A, Initial Issue

This procedure supersedes unit specific procedures PT/1/A/4600/06A and PT/2/A/4600/06A. It is identical to these procedures with the exception of the following changes:

1. Step 3.1 - The required time was changed from 2 hours to a more accurate duration of 1 hour.

2. Step 6.3 - A new step was added to provide warning of Radiation Monitor (EMF) alarms.

3. Step 8.2 - The step was revised to ensure that the applicable unit is identified to Radiation Protection (previously "Health Physics") for posting.

4. Step 8.3 - The requirement for Shift Supervisor signoff was deleted. The applicable unit's Containment Entry Log is to be reviewed by the Test Coordinator instead.

5. Step 12.19 - The step was changed to specify that detectors be withdrawn after voltage plateaus are obtained, instead of being operated in SCAN and RECORD.

6 Step 12.21 - The notification of Operations as well as Radiation Protection was specified.

7. Step 12.22 - The reference to Data Book was revised to make it applicable to either unit.

8. General - PT/1/A/4600/06A had previously specified that strip chart traces be retained. This issue adopts retention of detector voltage plateau data via PAO Table, as specified by PT/2/A/4600/06A. This PAO data is enhanced, however, by addition of provisions to record the applied detector voltage coincident with corresponding detector response noted on the PAO table.

None of the above changes introduces a nuclear safety concern. The Incore Detector (ENA) System is at all times operated in accordance with OP/0/A/6150/07, "Incore Instrumentation". Additional measures are adopted to ensure that no personnel radiation exposure results from ENA detector operation.

PT/1/A/4150/28, Initial Issue

The purpose of this test is to set the feedwater heater levels such that optimum performance is obtained. The method for obtaining these levels involves changing the shell liquid levels. A change in the shell liquid level can cause damage to the heater when the level at the entrance to the sub-cooler is too low, resulting in a flashing condition. Before damage causing low shell liquid level is reached, the Drain Cooler Approach temperature will increase significantly. At this point during the test, the level will not be lowered further. Also, the shell liquid levels will not be lowered more than 5" below the normal liquid level, even if a sharp increase in the Drain Cooler Approach does not occur, preventing the possibility of overranging related equipment. This will also prevent heater damage in the event of an equipment malfunction. The feedwater heaters are not safety related. No unreviewed safety question is created.

PT/0/A/4400/22B, Change #26

The purpose of this procedure change is to allow for the Nuclear Service Water (RN) Flow Balance: Train B; the Nuclear Service Water Pump Train B Performance Test; and the RN Pump 1B flow versus pressure data acquisition to be performed simultaneously. The flow path line-up will be established using the RN Flow Balance. No unique valve line-ups or alignments are required beyond the scope of the normal flow balance in the combination of these two procedures. Pump Inservice Test (IWP) data will be taken while the pump is aligned per the flow balance.

The RN system is not degraded by these tests. The RN pumps are not operated outside the Design conditions. The performance of these tests do not make any train of RN inoperable. No unreviewed safety questions are created.

PT/0/A/4400/08B, Change #4

The purpose of this procedure change is to allow for the Nuclear Service Water (RN) Flow Balance: Train B; the Nuclear Service Water Pump Train B Performance Test; and the RN Pump 1B flow versus pressure data acquisition to be performed simultaneously. The flow path line-up will be established using the RN Flow Balance. No unique valve line-ups or alignments are required beyond the normal scope of the Flow Balance. Valves 1RN48B, 1RN50B, and 1RN51B will be deleted from the valve line-up because Unit 1 is in mode 5, and the Unit 1 Non-Essential Header is not being supplied.

Before the completion of the RN Flow Balance on Pump 1B, a flow versus pressure data profile will be recorded. This is new enclosure 13.8 of the Flow Balance. While in the RN Flow Balance, the ordinary valve line-up is

such that the flow conditions are near the maximum flow paths available to RN in the one pump/isolated train alignment. Thus, by opening the final Containment Spray Heat Exchanger, a flow path of two Containment Spray, two Component Cooling, and two Diesel Generator Engine Cooling Water Heat Exchangers, one Control Area Ventilation/Chilled Water chiller, and various other flows maximize the flow for that data point of the pump curve profile. By gradually closing off each of the flow paths, a flow versus pressure profile may be constructed. Cautions to avoid the pump runout condition of 26,000 gpm and mini-flow dead heading of 8,500 gpm are emphasized throughout the procedure steps.

One concern beyond the scope of the RN Flow Balance is the possibility of a second pump start while the flow paths have been reduced in the gathering of low flow, high pressure data. On the possibility of a low pit level, a signal for the second train pump to start could exceed the immediate mini-flow alignment. The operators at the controls have the capability to open the Containment Spray Heat Exchangers from the control board. The Test Coordinator has the capability of removing air from 1RN351 and 2RN351, immediately allowing full flow to the Component Cooling Heat Exchangers. The Component Cooling Heat Exchangers would then be returned to their flow balance, fail safe positions. These cautions are also highlighted within the procedure. If the failure of the instrument air system occurred, the Component Cooling Heat Exchangers would be aligned to their fail safe position, removing any concerns of inadequate mini-flow protection for the two pumps. The Component Cooling Heat Exchangers can adequately provide a flow path to ensure no dead heading occurs for both pumps in operation.

The RN system is not degraded by these tests. The RN pumps are not operated outside the design conditions. The performance of these tests do not make any train of RN inoperable. No unreviewed safety questions are created.

TN/1/A/1005/02/26A, Change #1

Nuclear Station Modification (NSM) CN-11005/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the removal of snubbers deleted from Math Model NV-01. This change adds an alternate support 1-A-KC-4946 to the scope of this procedure. This support will be deleted from the Component Cooling (KC) system. In addition, this change provides direction allowing insulation steps to be NA'd if not required and properly documented within the work request.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems' piping during implementation of this procedure. Math Model NV-01 and the KC system have been qualified for the present support/restraint (S/R) configuration. They have also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined that this work may be done while the affected system(s) are operable, provided that all the support modifications for the entire math model are completed within

the 72 hours allowed by the Technical Specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified under any combination of snubbers removed, provided the modifications to the existing supports have been completed. This procedure is written to complete all modifications before deleting any supports. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated that they could justify operable status of any configuration of modified supports provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all Technical Specification requirements.

System isolations are not required, and controls are provided to ensure compliance with Technical Specifications. Post Modification Testing is not required for this modification. Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

PT/1/A/4400/01, Change #19

The Emergency Core Cooling System (ECCS) flow balance of the Safety Injection (NI) and Chemical and Volume Control (NV) system is normally performed in "No Mode" with the reactor vessel head removed and water allowed to overflow into the reactor cavity. This restricted procedure change will provide the necessary guidance to perform this balance with the reactor vessel head installed and the Reactor Coolant (NC) system filled and vented.

Chapter 15 of the FSAR addresses dilution accidents in Mode 5 and recognizes that by having proper operation of the Boron Dilution Mitigation System (BDMS), there is adequate time for termination of the dilution source and injection of borated water to ensure that shutdown margin is maintained. This procedure change ensures that shutdown margin is maintained by ensuring proper boron concentration in the Refueling Water Storage Tank (FWST), which is the injection source. FWST temperature is also verified to be above 80°F to ensure shutdown margin is not compromised by the injection of this source of water. It also ensures that the BDMS system is operable as required by Tech. Specs. for Mode 5. Equipment required to terminate this event will be available.

The test flowpath of the NI and NV system will be identical to the designed flowpath, which is to take water from the FWST through the NC system and the pressurizer, with venting through the Pressurizer Relief Tank (PRT). Proper termination criteria for pressurizer level is provided in the procedure, and the PRT is vented to lower containment through vent valves and directed to the containment auxiliary charcoal filters.

Operation of all pumps will be within their design requirements. By limiting the temperature difference between the FWST water and the NC system, Pressurizer and vessel heatup and cooldown limits are maintained within Tech. Spec. limits. FWST water quality is maintained within Tech. Spec. limits to

provide necessary makeup to the NC system. Proper termination criteria for pump parameters, temperature limits, and pressurizer level are provided in the test.

With the unit in Mode 5, there are no requirements by Tech. Specs. for the operation of the ECCS equipment. Only one NV pump is required to be operable with an assured source of emergency power. During this entire test, an operable boration flowpath, as defined by Tech. Specs., will be maintained. Also, the Residual Heat Removal (ND) train providing core cooling will be available for immediate operation as defined by Tech. Specs. This Tech. Spec. allows for the termination of ND flow for up to an hour, as long as no dilutions are in progress, and the NC system is below saturation by 10°F. The test provides termination criteria by limiting time that the ND pump can be off to 1 hour and ensuring the NC system temperature, as measured by core exit thermocouples, is not allowed to exceed 140°F, which is well below saturation temperature of 212°F at atmospheric pressure. As defined by Tech. Specs. for an NC system with filled loops, either another ND train will be operable, or two Steam Generators will be filled to greater than 12% Narrow Range Level.

All plant equipment will be operated within its design requirements as specified by the limits and precautions in the procedure. Pump protection is provided by insuring proper net positive suction head prior to pump start. A minimum flowpath is also verified prior to starting any pump. The flowpaths provided in the procedure do not differ from the designed flowpath of the operated systems.

The change ensures that all Tech. Spec. requirements are maintained by ensuring shutdown margin is maintained, a boron flowpath is maintained, and BDMS is available with backup from the source range channels. Core cooling requirements are maintained as specified for mode 5. Pressurizer and NC system heatup and cooldown limits are maintained. Therefore, no Unreviewed Safety Question exists by the use of this procedure change.

OP/1/A/6700/01, Change #200

Full Power Range (P/R) Nuclear Instrumentation System (NIS) Currents

This section of Table 2.2 contains P/R NIS full power calibration currents at axial offsets of +20%, 0%, and -20% derived per approved procedure PT/1/A/4600/05E, Preliminary NIS Calibration. The M Factors denoted in this section have not been changed (they remain the same as the factors determined per the final Cycle 5 performance of the quarterly Incore/Excore calibration). The new calibration currents are generated by adjusting the final calibration currents of Cycle 5 for the expected change in core power distribution from End of Cycle 5 to Beginning of Cycle 6. These currents are used by Instrument and Electrical to enhance the calibration of the NIS to ensure the most accurate possible indication of Axial Flux Difference (and f_1 (ΔI) input function to the Overtemperature Differential Temperature setpoints in the 7300 process circuitry) during initial power escalation to 30% Full Power (F.P.) (or the highest power level allowed per PT/1/A/4150/21, Controlling Procedure for Startup Testing) following refueling.

Power Range NIS Trip Setpoints

Trip setpoints for the P/R NIS Detectors may at no time be deliberately established at greater than 109% F.P. The trip setpoints may be set lower than 109% by approved procedure, Tech. Spec. Action statement, or by direction of the Shift Supervisor. The value of 109% F.P. is set forth by Tech. Specs. to ensure that power operation is at all times bounded by the assumptions used in FSAR Chapter 15 analyses. Any setpoints below 109% are for desired conservatism or compliance with Tech. Specs. In this case, the trip setpoints are established at 25% F.P. per Tech. Spec. Special Test Exception 3.10.3 and PT/1/A/4150/21, Controlling Procedure for Startup Testing, for the purpose of imposing strict conservatism on initial operation at below 10% F.P. following refueling. This ensures that all low power (particularly zero power physics) testing is satisfactorily completed, thereby validating the new core's design predictions, before continued power escalation is permitted.

No unreviewed safety questions are created by these changes.

OP/1/B/6100/10E, Change #12

Nuclear Station Modification CN-11168/00 has modified the Steam Generator (S/G) feedwater control system to use digital electronics. This system replaces the former feedwater control system's signal processing and output to include more redundancy of input, signal quality control, signal substitution during recognized control system malfunctions, and the transfer of feedwater control to the operator when automatic control is questionable. Components affected by the modification include the main feedwater valves and the main feedwater pumps. This evaluation covers only the procedure changes made to the annunciator responses that support the operation of the S/G digital feedwater control system (DFCS).

The annunciator responses are written to a specified format that precludes the need for discussion of the setpoint, the origin, the probable cause and the automatic actions. These items are determined by the modification, and reflect the as-built configuration of the plant. Immediate and supplemental actions are determined by knowledgeable individuals and bear review here.

"CFCV ISOL VLVS CLSD, E/5" has the operator determine which main feedwater isolation valve is closed and makes preparation to open it without causing a perturbation to the feedwater system. Subsequent action ensures the cause of the alarm is fully understood should the alarm not clear after the isolation valve is opened.

"DFCS NOT IN AUTO, D/5" has the operator manually control the feedwater system, since control of the system has been transferred from automatic. With the exception of one channel of feedwater flow, the operator has available all the indications available to the automatic control function. Manual control of the feedwater system is an analyzed and qualified operator function. Subsequent actions determine and correct the cause of the malfunction and restore the feedwater system to automatic control.

"DFCS TROUBLE, C/5" has the operator take appropriate actions to other DFCS alarms before requesting maintenance support in this response, since this alarm will actuate on any malfunction of the DFCS feedwater control system whether operator action is required or not. Of the four alarms discussed, this has lowest priority of response. Subsequent action is to determine and correct the cause of the malfunction and ensure the optimum performance of the control system.

"FCV OR BFCV DEVIATION, B/5" has the operator determine which feedwater or bypass control valve is experiencing a deviation between its demand position and its actual position. Based on plant conditions, actions are prescribed to either take manual control of the feedwater supply for the affected loop or monitor loop indications while the system corrects itself. Subsequent actions determine and correct the cause of the malfunction and describe the sequence for re-establishing automatic valve control.

The actions provided for operator control of the S/G feedwater control system in manual as a result of the modification, CN-11168/00, do not constitute an unreviewed safety question. Actions not previously prescribed are required to satisfy plant conditions or were previously accomplished by normal, abnormal or emergency operating procedures. Incorporation of these actions into this procedure merely reflect the addition of a mechanism to alert the operator to a need to operate manually or monitor the feedwater system for proper operation.

TN/2/B/0605/00/01A, Initial Issue

Nuclear Station Modification (NSM) CN-20605, Rev. 0, provides modifications to each main steam Atmospheric Dump Valve (ADV) that will improve the reliability and maintainability of these valves. Several of the valve components will either be replaced with a new component, or the existing one will be modified. These modifications will reduce the chances of valve failure and enhance the performance of the valve. The purpose of this procedure is to provide guidance for the modification to ADV 2SV030.

The steam dump system is not essential for safe shutdown of the unit; thus, it is not designated as safety related. The steam dump system merely provides added flexibility in unit operation. Per the Operations Group, one or two ADV's may be taken out of service while the unit is at 100% power. Implementing this procedure will require isolation of valve 2SV030. The Operations Group will coordinate the isolations necessary to implement this procedure. The isolations required for implementation of this procedure are the same as the isolations required when performing repair or maintenance on 2SV030.

Testing for NSM CN-20605, Rev. 0, will be performed in accordance with the Post Modification Testing program at the station. The valve and its associated instrumentation will be calibrated, setup and checked out to ensure proper operation and indication. A visual inspection for leakage will be performed at operating temperature and pressure with the associated block valve open. Also, a test will be performed by the Performance Group to demonstrate the ability of 2SV030 to open under main steam pre[illegible] conditions. This test has its own 10CFR50.59 evaluation perfo[illegible] and

approved before the test is performed during implementation of this procedure. The test will be performed at 97% power to minimize the potential for a plant transient if the valve opens and fails to close.

All of the testing referenced in this evaluation is identified on the Post Modification Testing Plan and has been reviewed by the system engineer to ensure the testing identified adequately addresses the concerns of the post modification testing program.

The Operations Group will control the isolations required to modify 2SV030. Post modification testing will be performed to ensure the valve performs its intended function. The ADVs are not used for accident mitigation. They have no interface with any plant system used for accident mitigation. No new operating modes or characteristics are introduced by this procedure. Neither any setpoint, design limit, nor any operating parameter is affected.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/2/B/0605/00/02A, Initial Issue

Nuclear Station Modification (NSM) CN-20605, Rev. 0, provides modifications to each main steam Atmospheric Dump Valve (ADV) that will improve the reliability and maintainability of these valves. Several of the valve components will either be replaced with a new component, or the existing one will be modified. These modifications will reduce the chances of valve failure and enhance the performance of the valve. The purpose of this procedure is to provide guidance for the modification to ADV 2SV032.

The steam dump system is not essential for safe shutdown of the unit; thus, it is not designated as safety related. The steam dump system merely provides added flexibility in unit operation. Per the Operations Group, one or two ADV's may be taken out of service while the unit is at 100% power. Implementing this procedure will require isolation of valve 2SV032. The Operations Group will coordinate the isolations necessary to implement this procedure. The isolations required for implementation of this procedure are the same as the isolations required when performing repair or maintenance on 2SV032.

Testing for NSM CN-20605, Rev. 0, will be performed in accordance with the Post Modification Testing program at the station. The valve and its associated instrumentation will be calibrated, setup and checked out to ensure proper operation and indication. A visual inspection for leakage will be performed at operating temperature and pressure with the associated block valve open. Also, a test will be performed by the Performance Group to demonstrate the ability of 2SV032 to open under main steam pressure conditions. This test has its own 10CFR50.59 evaluation performed and approved before the test is performed during implementation of this procedure. The test will be performed at 97% power to minimize the potential for a plant transient if the valve opens and fails to close.

All of the testing referenced in this evaluation is identified on the Post Modification Testing Plan and has been reviewed by the system engineer to

ensure the testing identified adequately addresses the concerns of the post modification testing program.

The Operations Group will control the isolations required to modify 2SV032. Post modification testing will be performed to ensure the valve performs its intended function. The ADVs are not used for accident mitigation. They have no interface with any plant system used for accident mitigation. No new operating modes or characteristics are introduced by this procedure. Neither any setpoint, design limit, nor any operating parameter is affected.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/2/B/0605/00/03A, Initial Issue

Nuclear Station Modification (NSM) CN-20605, Rev. 0, provides modifications to each main steam Atmospheric Dump Valve (ADV) that will improve the reliability and maintainability of these valves. Several of the valve components will either be replaced with a new component, or the existing one will be modified. These modifications will reduce the chances of valve failure and enhance the performance of the valve. The purpose of this procedure is to provide guidance for the modification to ADV 2SV034.

The steam dump system is not essential for safe shutdown of the unit; thus, it is not designated as safety related. The steam dump system merely provides added flexibility in unit operation. Per the Operations Group, one or two ADV's may be taken out of service while the unit is at 100% power. Implementing this procedure will require isolation of valve 2SV034. The Operations Group will coordinate the isolations necessary to implement this procedure. The isolations required for implementation of this procedure are the same as the isolations required when performing repair or maintenance on 2SV034.

Testing for NSM CN-20605, Rev. 0, will be performed in accordance with the Post Modification Testing program at the station. The valve and its associated instrumentation will be calibrated, setup and checked out to ensure proper operation and indication. A visual inspection for leakage will be performed at operating temperature and pressure with the associated block valve open. Also, a test will be performed by the Performance Group to demonstrate the ability of 2SV034 to open under main steam pressure conditions. This test has its own 10CFR50.59 evaluation per [illegible]ed and approved before the test is performed during implementation of this procedure. The test will be performed at 97% power to minimize the potential for a plant transient if the valve opens and fails to close.

All of the testing referenced in this evaluation is identified on the Post Modification Testing Plan and has been reviewed by the system engineer to ensure the testing identified adequately addresses the concerns of the post modification testing program.

The Operations Group will control the isolations required to modify 2SV034. Post modification testing will be performed to ensure the valve performs its intended function. The ADVs are not used for accident mitigation. They have

no interface with any plant system used for accident mitigation. No new operating modes or characteristics are introduced by this procedure. Neither any setpoint, design limit, nor any operating parameter is affected.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/2/B/0605/00/04A, Initial Issue

Nuclear Station Modification (NSM) CN-20605, Rev. 0, provides modifications to each main steam Atmospheric Dump Valve (ADV) that will improve the reliability and maintainability of these valves. Several of the valve components will either be replaced with a new component, or the existing one will be modified. These modifications will reduce the chances of valve failure and enhance the performance of the valve. The purpose of this procedure is to provide guidance for the modification to ADV 2SV036.

The steam dump system is not essential for safe shutdown of the unit; thus, it is not designated as safety related. The steam dump system merely provides added flexibility in unit operation. Per the Operations Group, one or two ADV's may be taken out of service while the unit is at 100% power. Implementing this procedure will require isolation of valve 2SV036. The Operations Group will coordinate the isolations necessary to implement this procedure. The isolations required for implementation of this procedure are the same as the isolations required when performing repair or maintenance on 2SV036.

Testing for NSM CN-20605, Rev. 0, will be performed in accordance with the Post Modification Testing program at the station. The valve and its associated instrumentation will be calibrated, setup and checked out to ensure proper operation and indication. A visual inspection for leakage will be performed at operating temperature and pressure with the associated block valve open. Also, a test will be performed by the Performance Group to demonstrate the ability of 2SV036 to open under main steam pressure conditions. This test has its own 10CFR50.59 evaluation performed and approved before the test is performed during implementation of this procedure. The test will be performed at 97% power to minimize the potential for a plant transient if the valve opens and fails to close.

All of the testing referenced in this evaluation is identified on the Post Modification Testing Plan and has been reviewed by the system engineer to ensure the testing identified adequately addresses the concerns of the post modification testing program.

The Operations Group will control the isolations required to modify 2SV036. Post modification testing will be performed to ensure the valve performs its intended function. The ADVs are not used for accident mitigation. They have no interface with any plant system used for accident mitigation. No new operating modes or characteristics are introduced by this procedure. Neither any setpoint, design limit, nor any operating parameter is affected.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/2/B/0605/00/05A, Initial Issue

Nuclear Station Modification (NSM) CN-20605, Rev. 0, provides modifications to each main steam Atmospheric Dump Valve (ADV) that will improve the reliability and maintainability of these valves. Several of the valve components will either be replaced with a new component, or the existing one will be modified. These modifications will reduce the chances of valve failure and enhance the performance of the valve. The purpose of this procedure is to provide guidance for the modification to ADV 2SV038.

The steam dump system is not essential for safe shutdown of the unit; thus it is not designated as safety related. The steam dump system merely provides added flexibility in unit operation. Per the Operations Group, one or two ADV's may be taken out of service while the unit is at 100% power. Implementing this procedure will require isolation of valve 2SV038. The Operations Group will coordinate the isolations necessary to implement this procedure. The isolations required for implementation of this procedure are the same as the isolations required when performing repair or maintenance on 2SV038.

Testing for NSM CN-20605, Rev. 0, will be performed in accordance with the Post Modification Testing program at the station. The valve and its associated instrumentation will be calibrated, setup and checked out to ensure proper operation and indication. A visual inspection for leakage will be performed at operating temperature and pressure with the associated block valve open. Also, a test will be performed by the Performance Group to demonstrate the ability of 2SV038 to open under main steam pressure conditions. This test has its own 10CFR50.59 evaluation performed and approved before the test is performed during implementation of this procedure. The test will be performed at 97% power to minimize the potential for a plant transient if the valve opens and fails to close.

All of the testing referenced in this evaluation is identified on the Post Modification Testing Plan and has been reviewed by the system engineer to ensure the testing identified adequately addresses the concerns of the post modification testing program.

The Operations Group will control the isolations required to modify 2SV038. Post modification testing will be performed to ensure the valve performs its intended function. The ADVs are not used for accident mitigation. They have no interface with any plant system used for accident mitigation. No new operating modes or characteristics are introduced by this procedure. Neither any setpoint, design limit, nor any operating parameter is affected.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/2/B/0605/00/06A, Initial Issue

Nuclear Station Modification (NSM) CN-20605, Rev. 0, provides modifications to each main steam Atmospheric Dump Valve (ADV) that will improve the reliability and maintainability of these valves. Several of the valve components will either be replaced with a new component, or the existing one will be modified.

These modifications will reduce the chances of valve failure and enhance the performance of the valve. The purpose of this procedure is to provide guidance for the modification to ADV 2SV040.

The steam dump system is not essential for safe shutdown of the unit; thus, it is not designated as safety related. The steam dump system merely provides added flexibility in unit operation. Per the Operations Group, one or two ADV's may be taken out of service while the unit is at 100% power. Implementing this procedure will require isolation of valve 2SV040. The Operations Group will coordinate the isolations necessary to implement this procedure. The isolations required for implementation of this procedure are the same as the isolations required when performing repair or maintenance on 2SV040.

Testing for NSM CN-20605, Rev. 0, will be performed in accordance with the Post Modification Testing program at the station. The valve and its associated instrumentation will be calibrated, setup and checked out to ensure proper operation and indication. A visual inspection for leakage will be performed at operating temperature and pressure with the associated block valve open. Also, a test will be performed by the Performance Group to demonstrate the ability of 2SV040 to open under main steam pressure conditions. This test has its own 10CFR50.59 evaluation performed and approved before the test is performed during implementation of this procedure. The test will be performed at 97% power to minimize the potential for a plant transient if the valve opens and fails to close.

All of the testing referenced in this evaluation is identified on the Post Modification Testing Plan and has been reviewed by the system engineer to ensure the testing identified adequately addresses the concerns of the post modification testing program.

The Operations Group will control the isolations required to modify 2SV040. Post modification testing will be performed to ensure the valve performs its intended function. The ADVs are not used for accident mitigation. They have no interface with any plant system used for accident mitigation. No new operating modes or characteristics are introduced by this procedure. Neither any setpoint, design limit, nor any operating parameter is affected.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/2/J/0605/00/07A, Initial Issue

Nuclear Station Modification (NSM) CN-20605, Rev. 0, provides modifications to each main steam Atmospheric Dump Valve (ADV) that will improve the reliability and maintainability of these valves. Several of the valve components will either be replaced with a new component, or the existing one will be modified. These modifications will reduce the chances of valve failure and enhance the performance of the valve. The purpose of this procedure is to provide guidance for the modification to ADV 2SV042.

The steam dump system is not essential for safe shutdown of the unit; thus, it is not designated as safety related. The steam dump system merely provides

added flexibility in unit operation. Per the Operations Group, one or two ADV's may be taken out of service while the unit is at 100% power. Implementing this procedure will require isolation of valve 2SV042. The Operations Group will coordinate the isolations necessary to implement this procedure. The isolations required for implementation of this procedure are the same as the isolations required when performing repair or maintenance on 2SV042.

Testing for NSM CN-20605, Rev. 0, will be performed in accordance with the Post Modification Testing program at the station. The valve and its associated instrumentation will be calibrated, setup and checked out to ensure proper operation and indication. A visual inspection for leakage will be performed at operating temperature and pressure with the associated block valve open. Also, a test will be performed by the Performance Group to demonstrate the ability of 2SV042 to open under main steam pressure conditions. This test has its own 10CFR50.59 evaluation performed and approved before the test is performed during implementation of this procedure. The test will be performed at 97% power to minimize the potential for a plant transient if the valve opens and fails to close.

All of the testing referenced in this evaluation is identified on the Post Modification Testing Plan and has been reviewed by the system engineer to ensure the testing identified adequately addresses the concerns of the post modification testing program.

The Operations Group will control the isolations required to modify 2SV042. Post modification testing will be performed to ensure the valve performs its intended function. The ADVs are not used for accident mitigation. They have no interface with any plant system used for accident mitigation. No new operating modes or characteristics are introduced by this procedure. Neither any setpoint, design limit, nor any operating parameter is affected.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/2/B/0605/00/08A, Initial Issue

Nuclear Station Modification (NSM) CN-20605, Rev. 0, provides modifications to each main steam Atmospheric Dump Valve (ADV) that will improve the reliability and maintainability of these valves. Several of the valve components will either be replaced with a new component, or the existing one will be modified. These modifications will reduce the chances of valve failure and enhance the performance of the valve. The purpose of this procedure is to provide guidance for the modification to ADV 2SV044.

The steam dump system is not essential for safe shutdown of the unit; thus, it is not designated as safety related. The steam dump system merely provides added flexibility in unit operation. Per the Operations Group, one or two ADV's may be taken out of service while the unit is at 100% power. Implementing this procedure will require isolation of valve 2SV044. The Operations Group will coordinate the isolations necessary to implement this procedure. The isolations required for implementation of this procedure are

the same as the isolations required when performing repair or maintenance on 2SV044.

Testing for NSM CN-20605, Rev. 0, will be performed in accordance with the Post Modification Testing program at the station. The valve and its associated instrumentation will be calibrated, setup and checked out to ensure proper operation and indication. A visual inspection for leakage will be performed at operating temperature and pressure with the associated block valve open. Also, a test will be performed by the Performance Group to demonstrate the ability of 2SV044 to open under main steam pressure conditions. This test has its own 10CFR50.59 evaluation performed and approved before the test is performed during implementation of this procedure. The test will be performed at 97% power to minimize the potential for a plant transient if the valve opens and fails to close.

All of the testing referenced in this evaluation is identified on the Post Modification Testing Plan and has been reviewed by the system engineer to ensure the testing identified adequately addresses the concerns of the post modification testing program.

...erations Group will control the isolations required to modify 2SV044. ...dification testing will be performed to ensure the valve performs its ...d function. The ADVs are not used for accident mitigation. They have ...rface with any plant system used for accident mitigation. No new ...ng modes or characteristics are introduced by this procedure. Neither ...tpoint, design limit, nor any operating parameter is affected.

...ed on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/2/B/0605/00/09A, Initial Issue

Nuclear Station Modification (NSM) CN-20605, Rev. 0, provides modifications to each main steam Atmospheric Dump Valve (ADV) that will improve the reliability and maintainability of these valves. Several of the valve components will either be replaced with a new component, or the existing one will be modified. These modifications will reduce the chances of valve failure and enhance the performance of the valve. The purpose of this procedure is to provide guidance for the modification to ADV 2SV054.

The steam dump system is not essential for safe shutdown of the unit; thus, it is not designated as safety related. The steam dump system merely provides added flexibility in unit operation. Per the Operations Group, one or two ADV's may be taken out of service while the unit is at 100% power. Implementing this procedure will require isolation of valve 2SV054. The Operations Group will coordinate the isolations necessary to implement this procedure. The isolations required for implementation of this procedure are the same as the isolations required when performing repair or maintenance on 2SV054.

Testing for NSM CN-20605, Rev. 0, will be performed in accordance with the Post Modification Testing program at the station. The valve and its associated instrumentation will be calibrated, setup and checked out to ensure

proper operation and indication. A visual inspection for leakage will be performed at operating temperature and pressure with the associated block valve open. Also, a test will be performed by the Performance Group to demonstrate the ability of 2SV054 to open under main steam pressure conditions. This test has its own 10CFR50.59 evaluation performed and approved before the test is performed during implementation of this procedure. The test will be performed at 97% power to minimize the potential for a plant transient if the valve opens and fails to close.

All of the testing referenced in this evaluation is identified on the Post Modification Testing Plan and has been reviewed by the system engineer to ensure the testing identified adequately addresses the concerns of the post modification testing program.

The Operations Group will control the isolations required to modify 2SV054. Post modification testing will be performed to ensure the valve performs its intended function. The ADVs are not used for accident mitigation. They have no interface with any plant system used for accident mitigation. No new operating modes or characteristics are introduced by this procedure. Neither any setpoint, design limit, nor any operating parameter is affected.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TT/2/B/9200/063, Initial Issue

This procedure will stroke the Atmospheric Dump Valves against differential pressure to verify proper operation of the modifications performed in CN-20605.

The valve stroke test will open one of the Atmospheric Dump Valves 25%. The procedure requires that the plant to be at or below 97% power due to the slight increase in power that will occur when the valve is opened.

The test does not configure the plant so that equipment operation will be jeopardized. If the atmospheric dump valve was to remain open or fail open, the Control Room Operator can isolate the steam leak by closing the associated block valve. The Atmospheric Dump Valves are considered to be non-safety related equipment and are not required for safe shutdown of the reactor. Since the valves are being operated as designed, no new accident is created that is not already addressed in the FSAR.

The valves will be tested within their designed function and shall not be challenged beyond their normal function. Even though the valve is non-safety, if it were to malfunction, the control room can close the block valve isolating the valve. This does not affect safety equipment.

The Atmospheric Dump Valves are not addressed by Tech. Specs.

No unreviewed safety question exists for this procedure.

TN/1/B/1224/00/01A, Initial Issue

Nuclear Station Modification (NSM) CN-11224, Rev. 0, provides modifications to each main steam Atmospheric Dump Valve (ADV) that will improve the reliability and maintainability of these valves. Several of the valve components will either be replaced with a new component, or the existing one will be modified. These modifications will reduce the chances of valve failure and enhance the performance of the valve. The purpose of this procedure is to provide guidance for the modification to ADV 1SV030.

The steam dump system is not essential for safe shutdown of the unit; thus, it is not designated as safety related. The steam dump system merely provides added flexibility in unit operation. Per the Operations Group, one or two ADV's may be taken out of service while the unit is at 100% power. Implementing this procedure will require isolation of valve 1SV030. The Operations Group will coordinate the isolations necessary to implement this procedure. The isolations required for implementation of this procedure are the same as the isolations required when performing repair or maintenance on 1SV030.

Testing for NSM CN-11224, Rev. 0, will be performed in accordance with the Post Modification Testing program at the station. The valve and its associated instrumentation will be calibrated, setup and checked out to ensure proper operation and indication. A visual inspection for leakage will be performed at operating temperature and pressure with the associated block valve open. Also, a test will be performed by the Performance Group to demonstrate the ability of 1SV030 to open under main steam pressure conditions. This test has its own 10CFR50.59 evaluation performed and approved before the test is performed during implementation of this procedure. The test will be performed at 97% power to minimize the potential for a plant transient if the valve opens and fails to close.

All of the testing referenced in this evaluation is identified on the Post Modification Testing Plan and has been reviewed by the system engineer to ensure the testing identified adequately addresses the concerns of the post modification testing program.

The Operations Group will control the isolations required to modify 1SV030. Post modification testing will be performed to ensure the valve performs its intended function. The ADVs are not used for accident mitigation. They have no interface with any plant system used for accident mitigation. No new operating modes or characteristics are introduced by this procedure. Neither any setpoint, design limit, nor any operating parameter is affected.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/B/1224/00/02A, Initial Issue

Nuclear Station Modification (NSM) CN-11224, Rev. 0, provides modifications to each main steam Atmospheric Dump Valve (ADV) that will improve the reliability and maintainability of these valves. Several of the valve components will either be replaced with a new component, or the existing one will be modified.

These modifications will reduce the chances of valve failure and enhance the performance of the valve. The purpose of this procedure is to provide guidance for the modification to ADV 1SV032.

The steam dump system is not essential for safe shutdown of the unit; thus, it is not designated as safety related. The steam dump system merely provides added flexibility in unit operation. Per the Operations Group, one or two ADV's may be taken out of service while the unit is at 100% power. Implementing this procedure will require isolation of valve 1SV032. The Operations Group will coordinate the isolations necessary to implement this procedure. The isolations required for implementation of this procedure are the same as the isolations required when performing repair or maintenance on 1SV032.

Testing for NSM CN-11224, Rev. 0, will be performed in accordance with the Post Modification Testing program at the station. The valve and its associated instrumentation will be calibrated, setup and checked out to ensure proper operation and indication. A visual inspection for leakage will be performed at operating temperature and pressure with the associated block valve open. Also, a test will be performed by the Performance Group to demonstrate the ability of 1SV032 to open under main steam pressure conditions. This test has its own 10CFR50.59 evaluation performed and approved before the test is performed during implementation of this procedure. The test will be performed at 97% power to minimize the potential for a plant transient if the valve opens and fails to close.

All of the testing referenced in this evaluation is identified on the Post Modification Testing Plan and has been reviewed by the system engineer to ensure the testing identified adequately addresses the concerns of the post modification testing program.

The Operations Group will control the isolations required to modify 1SV032. Post modification testing will be performed to ensure the valve performs its intended function. The ADVs are not used for accident mitigation. They have no interface with any plant system used for accident mitigation. No new operating modes or characteristics are introduced by this procedure. Neither any setpoint, design limit, nor any operating parameter is affected.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/B/1224/00/03A, Initial Issue

Nuclear Station Modification (NSM) CN-11224, Rev. 0, provides modifications to each main steam Atmospheric Dump Valve (ADV) that will improve the reliability and maintainability of these valves. Several of the valve components will either be replaced with a new component, or the existing one will be modified. These modifications will reduce the chances of valve failure and enhance the performance of the valve. The purpose of this procedure is to provide guidance for the modification to ADV 1SV034.

The steam dump system is not essential for safe shutdown of the unit; thus, it is not designated as safety related. The steam dump system merely provides

added flexibility in unit operation. Per the Operations Group, one or two ADV's may be taken out of service while the unit is at 100% power. Implementing this procedure will require isolation of valve 1SV034. The Operations Group will coordinate the isolations necessary to implement this procedure. The isolations required for implementation of this procedure are the same as the isolations required when performing repair or maintenance on 1SV034.

Testing for NSM CN-11224, Rev. 0, will be performed in accordance with the Post Modification Testing program at the station. The valve and its associated instrumentation will be calibrated, setup and checked out to ensure proper operation and indication. A visual inspection for leakage will be performed at operating temperature and pressure with the associated block valve open. Also, a test will be performed by the Performance Group to demonstrate the ability of 1SV034 to open under main steam pressure conditions. This test has its own 10CFR50.59 evaluation performed and approved before the test is performed during implementation of this procedure. The test will be performed at 97% power to minimize the potential for a plant transient if the valve opens and fails to close.

All of the testing referenced in this evaluation is identified on the Post Modification Testing Plan and has been reviewed by the system engineer to ensure the testing identified adequately addresses the concerns of the post modification testing program.

The Operations Group will control the isolations required to modify 1SV034. Post modification testing will be performed to ensure the valve performs its intended function. The ADVs are not used for accident mitigation. They have no interface with any plant system used for accident mitigation. No new operating modes or characteristics are introduced by this procedure. Neither any setpoint, design limit, nor any operating parameter is affected.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/B/1224/00/04A, Initial Issue

Nuclear Station Modification (NSM) CN-11224, Rev. 0, provides modifications to each main steam Atmospheric Dump Valve (ADV) that will improve the reliability and maintainability of these valves. Several of the valve components will either be replaced with a new component, or the existing one will be modified. These modifications will reduce the chances of valve failure and enhance the performance of the valve. The purpose of this procedure is to provide guidance for the modification to ADV 1SV036.

The steam dump system is not essential for safe shutdown of the unit; thus, it is not designated as safety related. The steam dump system merely provides added flexibility in unit operation. Per the Operations Group, one or two ADV's may be taken out of service while the unit is at 100% power. Implementing this procedure will require isolation of valve 1SV036. The Operations Group will coordinate the isolations necessary to implement this procedure. The isolations required for implementation of this procedure are

Storage Tank (FWST). The reactor vessel is open with no fuel in the core during performance of this test. NI and NV pumps discharge into the reactor vessel, and water is allowed to overflow into the reactor vessel cavity.

Performance of this test provides assurance that adequate Emergency Core Cooling System (ECCS) flows will be delivered to the Reactor Coolant System in the event of a Loss of Coolant Accident.

With the reactor vessel head removed, there is no possibility of overpressurizing the Reactor Coolant System. The Emergency Core Cooling System (ECCS) is not required to perform any safety function during "No Mode" operations.

Failure of the ECCS during this test would not result in any accident different than any already evaluated in the FSAR. The ECCS will be aligned in configurations similar to those which would be required following an accident condition. However, the reactor vessel will be open and FWST water will overflow into the cavity.

Technical Specifications do not require the ECCS to be operable during "No Mode" conditions. This test ensures that Technical Specification Surveillance Requirements are satisfied for the ECCS and that the system will be able to meet its design safety function when required during accident conditions.

Based on the above analysis, it is concluded that the proposed procedure does not involve an unreviewed safety question.

PT/1/A/4200/01N Retype, Changes 0 to 33 Incorporated

The changes made as a result of this retype are as follows:

1. Section 2.0 was completely revised to separately categorize the source documents. FSAR Section 6.3.4 was included as an additional reference source.

2. Section 5.0, step 5.1, changed "Sections 12.1 through 12.29" to "Enclosures 13.1 through 13.29". Steps 5.1.3 and 5.2.4, "Heise" was changed to "Pressure".

3. Section 6.0, was renumbered and steps 6.4, 6.5, 6.6, and 6.7 added. Step 6.4 assures the Test coordinator that a testing path exists. Step 6.5 makes aware the possibility of inaccurate reading based on the phenomenon of flashing. Step 6.6 is an administrative addition. Previous step 6.6 was deleted. It is incorporated into Section 2.0, Tech Spec 4.4.6.2.2.

4. Section 7.0 was renumbered, and step 7.1, "Sections 12.1 through 12.26" was changed to "Enclosures 13.1 through 13.29".

5. Section 8.0 was completely revised to group similar prerequisite conditions (Step 8.4) for Enclosures 13.1 through 13.29.

the same as the isolations required when performing repair or maintenance on 1SV036.

Testing for NSM CN-11224, Rev. 0, will be performed in accordance with the Post Modification Testing program at the station. The valve and its associated instrumentation will be calibrated, setup and checked out to ensure proper operation and indication. A visual inspection for leakage will be performed at operating temperature and pressure with the associated block valve open. Also, a test will be performed by the Performance Group to demonstrate the ability of 1SV036 to open under main steam pressure conditions. This test has its own 10CFR50.59 evaluation performed and approved before the test is performed during implementation of this procedure. The test will be performed at 97% power to minimize the potential for a plant transient if the valve opens and fails to close.

All of the testing referenced in this evaluation is identified on the Post Modification Testing Plan and has been reviewed by the system engineer to ensure the testing identified adequately addresses the concerns of the post modification testing program.

The Operations Group will control the isolations required to modify 1SV036. Post modification testing will be performed to ensure the valve performs its intended function. The ADVs are not used for accident mitigation. They have no interface with any plant system used for accident mitigation. No new operating modes or characteristics are introduced by this procedure. Neither any setpoint, design limit, nor any operating parameter is affected.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/B/1224/00/05A, Initial Issue

Nuclear Station Modification (NSM) CN-11224, Rev. 0, provides modifications to each main steam Atmospheric Dump Valve (ADV) that will improve the reliability and maintainability of these valves. Several of the valve components will either be replaced with a new component, or the existing one will be modified. These modifications will reduce the chances of valve failure and enhance the performance of the valve. The purpose of this procedure is to provide guidance for the modification to ADV 1SV038.

The steam dump system is not essential for safe shutdown of the unit; thus, it is not designated as safety related. The steam dump system merely provides added flexibility in unit operation. Per the Operations Group, one or two ADV's may be taken out of service while the unit is at 100% power. Implementing this procedure will require isolation of valve 1SV038. The Operations Group will coordinate the isolations necessary to implement this procedure. The isolations required for implementation of this procedure are the same as the isolations required when performing repair or maintenance on 1SV038.

Testing for NSM CN-11224, Rev. 0, will be performed in accordance with the Post Modification Testing program at the station. The valve and its associated instrumentation will be calibrated, setup and checked out to ensure

proper operation and indication. A visual inspection for leakage will be performed at operating temperature and pressure with the associated block valve open. Also, a test will be performed by the Performance Group to demonstrate the ability of 1SV038 to open under main steam pressure conditions. This test has its own 10CFR50.59 evaluation performed and approved before the test is performed during implementation of this procedure. The test will be performed at 97% power to minimize the potential for a plant transient if the valve opens and fails to close.

All of the testing referenced in this evaluation is identified on the Post Modification Testing Plan and has been reviewed by the system engineer to ensure the testing identified adequately addresses the concerns of the post modification testing program.

The Operations Group will control the isolations required to modify 1SV038. Post modification testing will be performed to ensure the valve performs its intended function. The ADVs are not used for accident mitigation. They have no interface with any plant system used for accident mitigation. No new operating modes or characteristics are introduced by this procedure. Neither any setpoint, design limit, nor any operating parameter is affected.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/B/1224/00/06A, Initial Issue

Nuclear Station Modification (NSM) CN-11224, Rev. 0, provides modifications to each main steam Atmospheric Dump Valve (ADV) that will improve the reliability and maintainability of these valves. Several of the valve components will either be replaced with a new component, or the existing one will be modified. These modifications will reduce the chances of valve failure and enhance the performance of the valve. The purpose of this procedure is to provide guidance for the modification to ADV 1SV040.

The steam dump system is not essential for safe shutdown of the unit; thus, it is not designated as safety related. The steam dump system merely provides added flexibility in unit operation. Per the Operations Group, one or two ADV's may be taken out of service while the unit is at 100% power. Implementing this procedure will require isolation of valve 1SV040. The Operations Group will coordinate the isolations necessary to implement this procedure. The isolations required for implementation of this procedure are the same as the isolations required when performing repair or maintenance on 1SV040.

Testing for NSM CN-11224, Rev. 0, will be performed in accordance with the Post Modification Testing program at the station. The valve and its associated instrumentation will be calibrated, setup and checked out to ensure proper operation and indication. A visual inspection for leakage will be performed at operating temperature and pressure with the associated block valve open. Also, a test will be performed by the Performance Group to demonstrate the ability of 1SV040 to open under main steam pressure conditions. This test has its own 10CFR50.59 evaluation performed and approved before the test is performed during implementation of this procedure.

The test will be performed at 97% power to minimize the potential for a plant transiant if the valve opens and fails to close.

All of the testing referenced in this evaluation is identified on the Post Modification Testing Plan and has been reviewed by the system engineer to ensure the testing identified adequately addresses the concerns of the post modification testing program.

The Operations Group will control the isolations required to modify 1SV040. Post modification testing will be performed to ensure the valve performs its intended function. The ADVs are not used for accident mitigation. They have no interface with any plant system used for accident mitigation. No new operating modes or characteristics are introduced by this procedure. Neither any setpoint, design limit, nor any operating parameter is affected.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/B/1224/00/07A, Initial Issue

Nuclear Station Modification (NSM) CN-11224, Rev. 0, provides modifications to each main steam Atmospheric Dump Valve (ADV) that will improve the reliability and maintainability of these valves. Several of the valve components will either be replaced with a new component, or the existing one will be modified. These modifications will reduce the chances of valve failure and enhance the performance of the valve. The purpose of this procedure is to provide guidance for the modification to ADV 1SV042.

The steam dump system is not essential for safe shutdown of the unit; thus, it is not designated as safety related. The steam dump system merely provides added flexibility in unit operation. Per the Operations Group, one or two ADV's may be taken out of service while the unit is at 100% power. Implementing this procedure will require isolation of valve 1SV042. The Operations Group will coordinate the isolations necessary to implement this procedure. The isolations required for implementation of this procedure are the same as the isolations required when performing repair or maintenance on 1SV042.

Testing for NSM CN-11224, Rev 0, will be performed in accordance with the Post Modification Testing program at the station. The valve and its associated instrumentation will be calibrated, setup and checked out to ensure proper operation and indication. A visual inspection for leakage will be performed at operating temperature and pressure with the associated block valve open. Also, a test will be performed by the Performance Group to demonstrate the ability of 1SV042 to open under main steam pressure conditions. This test has its own 10CFR50.59 evaluation performed and approved before the test is performed during implementation of this procedure. The test will be performed at 97% power to minimize the potential for a plant transient if the valve opens and fails to close.

All of the testing referenced in this evaluation is identified on the Post Modification Testing Plan and has been reviewed by the system engineer to

ensure the testing identified adequately addresses the concerns of the post modification testing program.

The Operations Group will control the isolations required to modify 1SV042. Post modification testing will be performed to ensure the valve performs its intended function. The ADVs are not used for accident mitigation. They have no interface with any plant system used for accident mitigation. No new operating modes or characteristics are introduced by this procedure. Neither any setpoint, design limit, nor any operating parameter is affected.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/B/1224/00/08A, Initial Issue

Nuclear Station Modification (NSM) CN-11224, Rev. 0, provides modifications to each main steam Atmospheric Dump Valve (ADV) that will improve the reliability and maintainability of these valves. Several of the valve components will either be replaced with a new component, or the existing one will be modified. These modifications will reduce the chances of valve failure and enhance the performance of the valve. The purpose of this procedure is to provide guidance for the modification to ADV 1SV044.

The steam dump system is not essential for safe shutdown of the unit; thus, it is not designated as safety related. The steam dump system merely provides added flexibility in unit operation. Per the Operations Group, one or two ADV's may be taken out of service while the unit is at 100% power. Implementing this procedure will require isolation of valve 1SV044. The Operations Group will coordinate the isolations necessary to implement this procedure. The isolations required for implementation of this procedure are the same as the isolations required when performing repair or maintenance on 1SV044.

Testing for NSM CN-11224, Rev. 0, will be performed in accordance with the Post Modification Testing program at the station. The valve and its associated instrumentation will be calibrated, setup and checked out to ensure proper operation and indication. A visual inspection for leakage will be performed at operating temperature and pressure with the associated block valve open. Also, a test will be performed by the Performance Group to demonstrate the ability of 1SV044 to open under main steam pressure conditions. This test has its own 10CFR50.59 evaluation performed and approved before the test is performed during implementation of this procedure. The test will be performed at 97% power to minimize the potential for a plant transient if the valve opens and fails to close.

All of the testing referenced in this evaluation is identified on the Post Modification Testing Plan and has been reviewed by the system engineer to ensure the testing identified adequately addresses the concerns of the post modification testing program.

The Operations Group will control the isolations required to modify 1SV044. Post modification testing will be performed to ensure the valve performs its intended function. The ADVs are not used for accident mitigation. They have

no interface with any plant system used for accident mitigation. No new operating modes or characteristics are introduced by this procedure. Neither any setpoint, design limit, nor any operating parameter is affected.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/B/1224/00/09A, Initial Issue

Nuclear Station Modification (NSM) CN-11224, Rev. 0, provides modifications to each main steam Atmospheric Dump Valve (ADV) that will improve the reliability and maintainability of these valves. Several of the valve components will either be replaced with a new component, or the existing one will be modified. These modifications will reduce the chances of valve failure and enhance the performance of the valve. The purpose of this procedure is to provide guidance for the modification to ADV 1SV054.

The steam dump system is not essential for safe shutdown of the unit; thus, it is not designated as safety related. The steam dump system merely provides added flexibility in unit operation. Per the Operations Group, one or two ADV's may be taken out of service while the unit is at 100% power. Implementing this procedure will require isolation of valve 1SV054. The Operations Group will coordinate the isolations necessary to implement this procedure. The isolations required for implementation of this procedure are the same as the isolations required when performing repair or maintenance on 1SV054.

Testing for NSM CN-11224, Rev. 0, will be performed in accordance with the Post Modification Testing program at the station. The valve and its associated instrumentation will be calibrated, setup and checked out to ensure proper operation and indication. A visual inspection for leakage will be performed at operating temperature and pressure with the associated block valve open. Also, a test will be performed by the Performance Group to demonstrate the ability of 1SV054 to open under main steam pressure conditions. This test has its own 10CFR50.59 evaluation performed and approved before the test is performed during implementation of this procedure. The test will be performed at 97% power to minimize the potential for a plant transient if the valve opens and fails to close.

All of the testing referenced in this evaluation is identified on the Post Modification Testing Plan and has been reviewed by the system engineer to ensure the testing identified adequately addresses the concerns of the post modification testing program.

The Operations Group will control the isolations required to modify 1SV054. Post modification testing will be performed to ensure the valve performs its intended function. The ADVs are not used for accident mitigation. They have no interface with any plant system used for accident mitigation. No new operating modes or characteristics are introduced by this procedure. Neither any setpoint, design limit, nor any operating parameter is affected.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

PT/1/A/4150/21 Retype, Changes 0 to 13 Incorporated

This reissue of Post Refueling Controlling Procedure for Startup Testing involves the following substantive changes:

- The evaluation of calculated R (using measured $F_{\Delta H}$) versus Reactor Coolant (NC) System Flow is no longer required due to revision of Tech. Spec. 3.2.3. Measured $F_{\Delta H}$ is no longer used to calculate P* (operating power level restriction) due to this Tech. Spec. revision as well. $F_{\Delta H}$ is now determined as a ratio of measured $F_{\Delta H}$ (inferred from measured F_Q) to $F_{\Delta H}$ limit (specified by the Core Operating Limits Report). Any restrictions on operating power level during power escalation are imposed by actions resulting from violations noted by this evaluation (per PT/1/A/4150/05, Core Power Distribution).

- Provisions for the performance of 10% Load Transient Tests at various power levels are added per the post Modification Testing Program for the Digital Feedwater Control System. This testing will be performed per TT/1/A/9200/062, and these evolutions will be under the control of approved operating procedures.

- The revision of the checkout of the Rod Control System Logic during approach to criticality allows for evaluation of the system modification, which now sets fully withdrawn (wd) position at 222 steps wd (with 110 step overlap). Manipulation of control rods is performed per Rod Control operating procedure and evaluation of required Shutdown Margin at this new control rod alignment has been performed by cognizant Design Engineering personnel.

- Provision to allow power escalation above 50% F.P. with an indicated Excore Tilt in excess of 1.02 has been added in accordance with revised Tech. Spec. 3.2.4. It is now permissible to waive compliance with this limit until the Incore/Excore Calibration test has been performed (at 76% power or 100% power) following refueling.

Revision of applicable Tech. Specs. was performed to comply with the new fuel vendor's (B & W) core design methodology. The subject changes ensure full compliance with the new Tech. Specs. Conservative approach to full power is assured by core monitoring (flux mapping) at the power levels specified in FSAR 14.3, Startup Physics Test Program. All restrictive actions imposed by peaking factor evaluation are observed throughout power escalation.

No unreviewed safety question created.

PT/1/A/4150/05 Retype, Changes 0 to 27 Incorporated

The changes incorporated by this reissue address compliance with the revision of the subject Tech. Specs., which have been changed in accordance with the core design methodology used by the new fuel vendor (B&W). These changes involve the following:

- Tech. Spec. 3/4.2.2 - The FQ-W(Z) evaluation for Baseload and RAOC modes of operation has been deleted from the surveillance requirements. The

Limiting Condition for Operation (LCO) is still verified using measured FQ (adjusted for measurement and design uncertainties). The new surveillance requirements involve evaluation of Operational and Reactor Protection System (RPS) Margins (the limits for which are specified in the cycle specific Core Operating Limits Report (COLR)). The Action Statements for LCO and Surveillance requirements involve forced reduction of Axial Flux Difference (AFD) operating space, allowable operating power level, K1 term of Overtemperature Differential Temperature (OTΔT), and K4 term of Overpower Differential Temperature (OPΔT). These penalties are imposed per changes incorporated in this procedure. Additionally, extrapolations of Operational and RPS Margins are evaluated to ensure that adequate margin to limit will exist for the next surveillance interval (31 Effective Full Power Days (EFPD)). If margin is projected to be lost during this period, the next flux map is required prior to the burnup (B/U) at which margin is lost, or a 2% penalty is imposed on measured FQ margin and appropriate punitive actions taken as required. Provisions for covering these contingencies are incorporated per this reissue, as well.

- Tech. Spec. 3/4.2.3 - The Calculated R vs Reactor Coolant (NC) System flow evaluation has been deleted from the LCO. The measured FΔH has been superseded by an evaluation of Measured Radial Peak Ratio to Allowable Radial Peak Ratio (both specified by the applicable COLR). Extrapolation of the margin between these ratios is performed (with subsequent 2% penalty on Ratio Margin or requirement for flux map prior to B/U at which extrapolated margin is lost) as is done for FQ. Punitive actions for margin violations include operating power level restrictions and reduction of K1 term of OTΔT, which are handled by incorporated changes.

The SNA Core incore computer code used to analyze raw flux map data, which provides output of measured FQ peaking factors, and the new MONITOR code (from B&W), which computes FΔH margins and performs extrapolations, have both been rigorously benchmarked and QA'd by cognizant B&W and Duke Power Design personnel. The method by which raw incore data is obtained (via Movable Incore Detector System) is unaffected by these changes.

These changes ensure full compliance with these Tech. Specs. at all times during cycle operation by either imposing the restrictions previously discussed or mandating more frequent core monitoring. No unreviewed safety questions are created by these changes.

TN/2 B/0623/00/01A, Initial Issue

This Procedure will add three backup filters on the Radiation Protection clothes dryers to prevent lint from clogging air flow monitoring device 2ABFX-AFMD-2. This is being done to ensure that the Auxiliary Building Ventilation (VA) System can carry out its function. Catawba Nuclear Station Technical Specifications 3/4.7.7 require that the flow rate of each train be tested and shown to be 30,000 CFM ±10%. The surveillance is performed with the use of air flow monitor device ABFX-AFMD-1 located upstream of the VA filtered exhaust subsystems, and air flow monitor ABFX-AFMD-2, located downstream of the VA exhaust filters used for flow balance testing.

During implementation of this procedure, the VA System will remain in service. The dryers will be isolated by red tagging Motor Control Center (MCC) SMXH breakers F06E, F06F and F06G OPEN. These breakers isolate only electrical power to the three Radiation Protection clothes dryers, and their isolation does not present a concern for safe plant operation.

Testing will be performed by electrically checking dryers ODRYR0001, ODRYR0002 and ODRYR0004 for correct operation and phase rotation, and by checking air flow monitor 2ABFX-AFMD-2 throughout the Unit 2 End of Cycle 4 outage to insure backup filters are performing as desired.

No failure modes are being introduced. The related aspects of the VA System will be maintained during the implementation of this procedure. Based on the above discussion, it is determined that an unreviewed safety question does not exist.

PT/2/A/4200/09A, Change #76

This change deletes the requirement for closing Containment Penetration Valve Injection Water (NW) System Valve 2NW-62, NW Surge Chamber 2B Nuclear Service Water (RN) Supply Isolation, from Enclosure 13.40A. This valve is closed for normal testing as a precautionary measure to reduce the chance of RN flow to the surge chamber. Closing the valve makes Train B NW inoperable. Train A NW is inoperable at this time, due to RN being isolated. Therefore, it is desirable to keep Train B NW operable. 2NW-61B must be tested from closed to open for operability. B Train NW pressure is higher than RN header pressure; there-fore, no flow to the NW Surge Chamber should occur. The inlet valve, 2NW-61B will be open for testing for less than 10 minutes.

Flow to surge chamber should not occur due to the pressure differential. No unreviewed safety question is created.

PT/2/A/4200/27, Change #13

The Control Room closed indication for Containment Penetration Valve Injection Water System (NW) valve 2NW-110B does not work and is to be repaired under Work Request 47898OPS. This work request is scheduled for the Unit 2 Trip List. Restricted change #13 to PT/2/A/4200/27 allows stroke time testing of valve 2NW-110B from closed to open without the green closed indication. The fail safe position of this valve is open. The Valve Inservice (IWV) test times the valve from initiation signal (depressing of open control room pushbutton) to open indication. Placing the valve in the closed position is accomplished by depressing the open control room pushbutton, verifying the open red light is illuminated and then depressing the close control room pushbutton and verifying the open red light is not illuminated.

Valve position is sensed by an open and a closed reed switch. Total stem travel is less than 0.5 inches, and the average stroke time is approximately 0.2 seconds. Position indication is either open or closed. There is no

"Intermediate" indication. Verifying, on depressing the close pushbutton, that the red open indication extinguishes confirms that the reed switch sensed a change of valve position.

A flow/no-flow test to prove valve closure would be impractical, as this would require entry into containment. Reed switches have been known to fail due to the heat generated if the valve is energized for an extended period of time. They have also come out of adjustment due to cycling of the valve.

Due to the knowledge that the valve is moving from the open position and because of the short travel and extremely fast stroke time, it is reasonable to assume that the valve is traveling to the closed position and thus the stroke time recorded under this modified test is valid.

Travel to the fail safe position of this valve is proven. The function of this valve is to open to provide seal water for 2KC424B. No unreviewed safety question is created.

TN/1/B/1197/00/01A, Initial Issue

This procedure will provide work activities necessary for the replacement of the High Temperature Cutout Switches on the Ice Condenser (NF) Air Handling Unit (AHU) Defrost Heaters for Unit 1.

High Temperature Cutout Switches on the Ice Condenser (NF) Air Handling Unit (AHU) Defrost Heaters for Unit 1 presently have to be manually reset if they trip. This creates problems because the switches are located in upper containment, and the trip condition is only observed during Operation's weekly walk through. Ice build-up due to the defrost heaters not operating when the switches are tripped is the final result. These switches protect the NF AHU heaters from an over temperature condition which could result in equipment damage and system inoperability. They are being replaced with automatic resetting switches which will perform the same function; however, they will reset automatically if temperature conditions return to normal.

The replacement of the old switches and the retest for the new switches will be controlled by this procedure. Replacement of the old switches will require the associated AHUs to be electrically isolated for personnel safety. Because this procedure can be implemented with Unit 1 operating (mode 1), the number of AHUs removed from service has to be controlled to ensure NF AHU cooling capacity is maintained. This procedure controls the number of AHUs removed from service at 2 AHUs or 4 switches at any one time. This will ensure that the cooling capacity, as stated in the design bases for the NF AHUs, is maintained. The new switches will be installed prior to the retest. All AHU timers will be reset to the correct time of day such that they will defrost at the correct time intervals set for normal operations. The heater trip circuitry will be fully retested using a heat gun as a heat catalyst for the new switches. This will prevent the heaters from being subjected to a large heat load while testing. All retesting will ensure that the new switches will perform all of their intended functions per the design bases. No other systems will be affected by this procedure.

This procedure will not create any unreviewed safety questions.

TN/2/A/0565/00/01A, Initial Issue

The referenced procedure will be used to provide guidance in the implementation and check out of Nuclear Station Modification (NSM) CN-20565. This modification will repower the Train A Boron Dilution Mitigation System (ENC), referred to as the Train A Neutron Flux Monitoring System in the scope document for NSM CN-20565. During normal plant operating conditions, power will be supplied from a 1E power source (2ERPA BKR #20), and during a Safe Shutdown Facility (SSF) event, power will be supplied from an SSF power source (SKPG BKR #36).

The implementation of this procedure will involve pulling nonsafety cable 2ENC539 and safety cables 2*ENC540 and 2*ENC541, along with picking up spare conductors from safety cables. Cables 2*ENC501, 2*ENC502, 2*ENC524 and 2*ENC529 will be voided. Fire watches will be established when and if fire boundaries are breached. This procedure will also involve mounting 2TBOX0603. The system and panels affected by the implementation of this procedure are safety related. Wiring installed in this procedure will effect systems and components that are energized. This procedure will be implemented in compliance with all applicable Technical Specifications. Limits, precautions, warnings, and notes are listed in the procedure such that systems important to safety are not degraded. The implementation of this procedure will not affect the bases for any of the systems as stated in the Tech. Specs.

Accordingly, this procedure will not create any unreviewed safety questions.

TN/2/A/0313/00/01A, Initial Issue

The Boric Acid Tank (BAT) does not have adequate overpressure protection. This problem could result in failure of the tank. The purpose of Nuclear Station Modification (NSM) CN-20313, Rev. 0, is to modify the below the diaphragm vent line so automatic venting capability is provided. The purpose of this procedure is to provide guidance for the modifications to the below the diaphragm vent line.

Implementing this procedure will require isolation and draining of the BAT to 90% level; therefore, the BAT will still be available as a source of borated water. The Operations Group will coordinate the isolations necessary to implement this procedure. The modification to the vent line may be performed during an outage in Modes 5, 6, and No Mode. The BAT will be out of service during the modification. The systems and equipment affected by this procedure can be out of service during Modes 5, 6, and No Mode.

Testing for NSM CN-20313, Rev. 0, will be performed in accordance with the Post Modification Testing program at the station. The new relief valve will be tested and verified to relieve at the design setpoint. This testing will be performed before and after the valve is installed in the system. The test of the valve before installation will verify the valve has been properly designed by the manufacturer. The test of the valve after installation will verify that the new test connection will perform its intended function, which is to provide a flowpath for in-line testing of the relief valve. The testing after installation of the relief valve will be performed in Modes 5, 6, or no mode while the tank is out of service. In addition, the BAT level protection

channels will be calibrated to verify proper operation of the instrumentation. Since the BAT is designed as an atmospheric tank, pressure testing of the new piping and components is not required.

The Operations Group will control the isolations required to implement this procedure. Post modification testing will be performed to ensure the relief valve performs its intended function. No new operating modes or characteristics are introduced by this procedure. Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

PT/2/A/4250/03B Retype, Changes 0 to 16 Incorporated

The intent of this procedure is to ensure the operational readiness of Motor Driven Auxiliary Feedwater (CA) Pump 2B, as well as to verify that the Valve Inservice Test (IWV) requirements for several critical CA check valves are satisfied.

Several changes included in this retype will ensure increased availability of CA Pump 2B. These changes include the addition of compensatory measures to this procedure, which will allow CA Pump 2B to be considered "Available" during the performance of this test. Likewise, the alignment of critical CA valves within the procedure will minimize the amount of time that CA Pump 2B is in recirculation alignment to the Upper Surge Tank, and unable to feed the Steam Generators without requiring Operator action. Several changes were included to allow the Operations monthly Head and Valve Verification Procedure to be performed in conjunction with this procedure, further reducing the amount of "Unavailable" time for CA Pump 2B.

The addition to this procedure of backflow testing requirements on the miniflow check valves for CA Pumps 2A and #2 ensures that the probability of malfunction of these check valves is decreased. Likewise, limits and precautions have been added as a result of this retype to ensure that bearing temperatures do not exceed a specified limit during this test, and that the time limit of miniflow operation is observed. These precautions will actually decrease the probability of a malfunction of CA Pump 2B.

A limit and precaution in this procedure has Operations verify that CA Pump 2B can be removed from service for the duration of the test. This will ensure that the two remaining CA Pumps are in service and able to perform their safety function before this test is allowed to proceed. Also, the addition of compensatory measures as described above will ensure that CA Pump 2B is able to perform its safety function with minimal operator action. This will serve as a further means of maintaining the margin of safety as described by Tech. Specs.

The remaining changes included in this retype that are not discussed above are administrative in nature. Therefore, no unreviewed safety questions exist as a result of this procedure retype.

PT/2/A/4250/03A Retype, Changes 0 to 19 Incorporated

The intent of this procedure is to ensure the operational readiness of Motor Driven Auxiliary Feedwater (CA) Pump 2A, as well as to verify that the Valve Inservice Test (IWV) requirements for several critical CA check valves are satisfied.

Several changes included in this retype will ensure increased availability of CA Pump 2A. These changes include the addition of compensatory measures to this procedure, which will allow CA Pump 2A to be considered "Available" during the performance of this test. Likewise, the alignment of critical CA valves within the procedure will minimize the amount of time that CA Pump 2A is in recirculation alignment to the Upper Surge Tank, and unable to feed the Steam Generators without requiring Operator action. Several changes were included to allow the Operations monthly Head and Valve Verification Procedure to be performed in conjunction with this procedure, further reducing the amount of "Unavailable" time for CA Pump 2A.

The addition to this procedure of backflow testing requirements on the mini-flow check valves for CA Pumps 2B and #2 ensures that the probability of malfunction of these check valves is decreased. Likewise, limits and precautions have been added as a result of this retype to ensure that bearing temperatures do not exceed a specified limit during this test, and that the time limit of miniflow operation is observed. These precautions will actually decrease the probability of a malfunction of CA Pump 2A.

A limit and precaution in this procedure has Operations verify that CA Pump 2A can be removed from service for the duration of the test. This will ensure that the two remaining CA Pumps are in service and able to perform their safety function before this test is allowed to proceed. Also, the addition of compensatory measures as described above will ensure that CA Pump 2A is able to perform its safety function with minimal operator action.

The remaining changes included in this retype that are not discussed above are administrative in nature. Therefore, no unreviewed safety questions exist as a result of this procedure retype.

MP/0/A/7600/108 Retype, Changes 0 to 1 Incorporated

This procedure is to perform corrective maintenance on valves used in various system applications. This procedure has been reviewed against approved vendor manuals, design documents, and station procedures to ensure that the corrective maintenance controlled by this procedure will return the valve to as-built/as-designed condition. These actions will ensure valve compliance with FSAR accident analysis. Therefore, no unreviewed safety question exists.

MP/0/A/7450/31 Retype #2, Changes 0 to 3 Incorporated

This procedure change is being done to give more detail and better guidance for obtaining charcoal samples from MSA carbon beds.

Tech. Spec. 3/4.7 is affected by this procedure. Operations has the responsibility and the procedures for compliance with this Tech. Spec. Test samples will be taken on these filter units when Tech. Specs. allow per Operations procedures. These changes will clarify and assure that the filter bed will be returned to the as-designed condition.

Activities performed under this procedure have been reviewed against approved vendor manuals, design documents and station procedures to ensure that activities controlled by this procedure will return the filter bed to the as-built/as-designed condition. These actions will ensure the filters' compliance with FSAR accident analysis. Therefore, no unreviewed safety question exists.

MP/0/A/7300/05, Initial Issue

This safety evaluation is for the rewrite of PM/IG-84 to MP/0/A/7300/05. The following is a summary of the changes made to this procedure.

- All sections were renumbered to fit the Maintenance Procedure format.
- Section 11.2 was added for the quarterly preventive maintenance of the turbine governor valve, force check of the governor valve, and oil pressure testing of the governor.
- Steps 11.2.3.18, 11.3.7 and 11.4.5.2 were added to obtain additional oil samples from the pump, turbine, and governor.

Tech. Specs. 3/4.7.1 is affected by this procedure. Operations has the responsibility and the procedures for compliance with these Tech. Specs. Maintenance will be performed on this pump/turbine when Tech. Specs. allow, per Operation's procedures. This rewrite will clarify and assure that maintenance activities will return the pump/turbine to as-designed conditions.

The changes made by this rewrite have been reviewed against approved vendor manuals, design documents, and station procedures to ensure that the preventive maintenance controlled by this procedure will return the pump/turbine to as-built/as-designed condition. These actions will ensure that the pump/turbine complies with FSAR accident analysis. Therefore, no unreviewed safety question exists.

PT/1/A/4150/28, Change #1

The test procedure involved in this change is used to optimize the feedwater heater levels and involves no safety related equipment. The change being made will delete the "Instrument Range" column from Enclosure 13.2. The deletion of this column will allow the Test Coordinator to select test equipment

specified in this procedure based on the needed rang and equipment availability. This change will in no way degrade the performance of this test. No unreviewed safety question is created.

TN/2/A/3342/CE/01A, Initial Issue

In order to comply with the ASME Code Section XI Inservice Inspection requirements, a radiographic examination must be performed on the weld located between Steam Generator (S/G) 2C and the 32" diameter main steam (SM) piping. In order to perform this radiography efficiently, a hole is needed in the SM system piping to allow access inside the piping. Exempt Change CE-3342 was originated to provide the access hole for the radiographic examination of the above weld. After the examination is complete, the hole will be plugged. The purpose of this procedure is to provide guidance for the drilling of the hole and the installation of the half coupling and plug.

The Operations Group will coordinate the isolations necessary to implement this procedure. S/G 2C and its associated main steam line will be out of service during the installation phase of this procedure. The drilling of the hole and the installation of the half coupling and plug will be performed during Modes 5, 6, and no mode. The Operations Group will be notified to ensure containment integrity is maintained while the affected main steam line is open to containment.

Testing for CE-3342 will be performed in accordance with the Post Modification Testing program at the station. Since the hole is one inch in diameter, a pressure test is not required per ASME Section XI exemptions; however, a visual inspection for leaks will be performed in Mode 3 with the affected SM piping at normal system temperature and pressure. This inspection will not affect the function or operation of the SM system. System isolations or abnormal alignments are not required.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/A/3283/CE/02A, Initial Issue

This procedure provides guidance for the implementation of Exempt Change CE-3283, Work Unit 02A. This modification provides manual throttling capability for Safety Injection (NI) valve 2NI178B. This will be accomplished by adding a third position to the power disconnect switch located on 2MC11, which will defeat the seal-in circuitry for the M/O and M/C circuits. This will allow 2NI178B to be positioned to any intermediate position by depressing the OPEN or CLOSE pushbutton, then releasing the pushbutton when the valve reaches the desired position. This capability will only be available when the disconnect switch is placed in the throttle (new) position.

All work in this procedure will be performed with Unit 2 in modes 5, 6, or No Mode and Residual Heat Removal (ND) Train B out of service. No plant equipment other than the valve will be affected. The changes described in this procedure involve only electrical wiring changes to the valve control circuits; no changes to the valve or valve actuator will be performed.

Electrical isolations involve red tagging specific breakers open. Testing involves stroking the valve full open and closed to verify that the present valve operating characteristics are maintained. The steps contained in Periodic Test PT/2/A/4200/18 which verify power removal for 2NI178B will be performed to verify that the ability to remove power to the valve actuator and comply with Technical Specification 3/4.5.2 has not been degraded. Since the purpose of this modification is to provide throttling capability, the valve will be placed in the throttle mode, and the control board pushbutton will be used to position the valve to various intermediate positions between full open and full closed. All testing will be completed prior to entering Mode 4 to ensure proper system operation before returning it to service.

This procedure will be performed when ND Train B is not required to be operable or perform a safety function within modes 5, 6, or No Mode. This procedure creates no new failure modes, and the failure analysis presented in the FSAR for this valve will not be affected. No operating parameters, safety limits, or setpoints will be changed.

There will be no Unreviewed Safety Questions created by this procedure.

TN/2/A/3283/CE/01A, Initial Issue

This procedure provides guidance for the implementation of Exempt Change CE-3283, Work Unit 01A. This modification provides manual throttling capability for Safety Injection (NI) valve 2NI173A. This will be accomplished by adding a third position to the power disconnect switch located on 2MC11, which will defeat the seal-in circuitry for the M/O and M/C circuits. This will allow 2NI173A to be positioned to any intermediate position by depressing the OPEN or CLOSE pushbutton, then releasing the pushbutton when the valve reaches the desired position. This capability will only be available when the disconnect switch is placed in the throttle (new) position.

All work in this procedure will be performed with Unit 2 in modes 5, 6, or No Mode and Residual Heat Removal (ND) Train A out of service. No plant equipment other than the valve will be affected. The changes described in this procedure involve only electrical wiring changes to the valve control circuits; no changes to the valve or valve actuator will be performed. Electrical isolations involve red tagging specific breakers open. Testing involves stroking the valve full open and closed to verify that the present valve operating characteristics are maintained. The steps contained in Periodic Test PT/2/A/4200/18 which verify power removal for 2NI173A will be performed to verify that the ability to remove power to the valve actuator and comply with Technical Specification 3/4.5.2, has not been degraded. Since the purpose of this modification is to provide throttling capability, the valve will be placed in the throttle mode, and the control board pushbutton will be used to position the valve to various intermediate positions between full open and full closed. All testing will be completed prior to entering Mode 4 to ensure proper system operation before returning it to service.

This procedure will be performed when ND Train A is not required to be operable or perform a safety function within modes 5, 6, or No Mode. This procedure creates no new failure modes, and the failure analysis presented in

the FSAR for this valve will not be affected. No operating parameters, safety limits, or setpoints will be changed.

No Unreviewed Safety Questions are created by this procedure.

TN/2/A/3097/CE/01A, Initial Issue

In order to comply with the ASME Code Section XI Inservice Inspection requirements, a radiographic examination must be performed on the weld located between Steam Generator (S/G) 2D and the 32" diameter main steam (SM) piping. In order to perform this radiography efficiently, a hole is needed in the SM system piping to allow access inside the piping. Exempt Change CE-3097 was originated to provide the access hole for the radiographic examination of the above weld. After the examination is complete, the hole will be plugged. The purpose of this procedure is to provide guidance for the drilling of the hole and the installation of the half coupling and plug.

The Operations Group will coordinate the isolations necessary to implement this procedure. S/G 2D and its associated main steam line will be out of service during the installation phase of this procedure. The drilling of the hole and the installation of the half coupling and plug will be performed during Modes 5, 6, and no mode. The Operations Group will be notified to ensure containment integrity is maintained while the affected main steam line is open to containment.

Testing for CE-3097 will be performed in accordance with the Post Modification Testing program at the station. Since the hole is one inch in diameter, a pressure test is not required per ASME Section XI exemptions; however, a visual inspection for leaks will be performed in Mode 3 with the affected SM piping at normal system temperature and pressure. This inspection will not affect the function or operation of the SM system. System isolations or abnormal alignments are not required.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/1/B/0758/00/01A, Initial Issue

This procedure will perform activities for Nuclear Station Modification (NSM) CN-10758, Rev. 0. This modification will upgrade the existing Recirculation Filtering System that services the Diesel Generator Engine Fuel Oil (FD) Storage Tanks. This upgraded filtration system will be a vendor supplied unit comprised of a clay polisher with a filter on its upstream and downstream side and a high capacity pump.

The implementation of this procedure will involve a concrete pour in the yard, pipe installation, removal of instruments, conduit installation, rerouting existing non-safety cable to the new vendor skid, relocating Motor Control Center (MCC) cubicles, and checkout of new piping and electrical system. All work performed by this procedure is non-safety and will not affect the operation of any safety related equipment. The modified portion of the FD System, located in the yard, involves Class G piping and is still isolated

from the safety portion by Class C normally closed valves and check valves. Electrical and instrument isolation will consist of opening MCC-1MXC Breaker R04G (Fuel Oil Recirculating Pump Motor 1PMTR0076) and closing the root valves for loop 1FD5030.

Provisions have been made with Operations and Chemistry personnel to assure we do not violate our requirements to check for and remove accumulated water from the fuel oil storage tanks once every 31 days per Tech. Spec. 4.8.1.1.2.C and obtain a sample of fuel oil for particulate contamination once every 31 days per Tech. Spec. 4.8.1.1.2.F.

The new system will operate as before, but will provide cleaner fuel to the Diesel Generators.

The modification is isolated by closed Class C valves and check valves from the safety related portions of the system. Also, the interlocks to shut down the recirculation pump and prevent adverse system affects on Diesel Generator operation are unchanged.

There are no unreviewed safety questions associated with this procedure.

TN/2/A/0564/00/01A, Initial Issue

This Nuclear Station Modification (NSM) modifies various meter scales, recorder scales, nameplates, pushbuttons, and switch escutcheons on the Unit 2 Main Control Board, the Unit 2 Auxiliary Feedwater Pump Turbine Control Panels, and Auxiliary Shutdown Panels A and B to conform with Human Factors Standards.

The implementation of this NSM will not cause the degradation of any plant safety system. The changes are mostly cosmetic in nature and take place on the surface of the above mentioned boards. This work will not cause the seismic characteristics of the boards to change in any manner. This work will be accomplished with the unit in Mode 4 and below.

No unreviewed safety question is created by this work.

TN/2/B/0103/00/01A, Initial Issue

This procedure will perform activities for Nuclear Station Modification (NSM) CN-20103, Rev. 0. This modification will upgrade the existing Recirculation Filtering System that services the Diesel Generator Engine Fuel Oil (FD) Storage Tanks. This upgraded filtration system will be a vendor supplied unit comprised of a clay polisher with a filter on its upstream and downstream side and a high capacity pump.

The implementation of this procedure will involve a concrete pour in the yard, pipe installation, removal of instruments, conduit installation, rerouting existing non-safety cable to the new vendor skid, relocating Motor Control Center (MCC) cubicles, and checkout of new piping and electrical system. All work performed by this procedure is non-safety and will not affect the operation of any safety related equipment. The modified portion of the FD

System, located in the yard, involves Class G piping and is still isolated from the safety portion by Class C normally closed valves and check valves. Electrical and instrument isolation will consist of opening MCC-2MXC Breaker R04G (Fuel Oil Recirculating Pump Motor 2PMTR0076) and closing the root valves for loop 2FD5030.

Provisions have been made with Operations and Chemistry personnel to assure we do not violate our requirements to check for and remove accumulated water from the fuel oil storage tanks once every 31 days per Tech. Spec. 4.8.1.1.2.C and obtain a sample of fuel oil for particulate contamination once every 31 days per Tech. Spec. 4.8.1.1.2.F.

The new system will operate as before, but will provide cleaner fuel to the Diesel Generators.

The modification is isolated by closed Class C valves and check valves from the safety related portions of the system. Also, the interlocks to shut down the recirculation pump and prevent adverse system affects on Diesel Generator operation are unchanged.

There are no unreviewed safety questions associated with this procedure.

TN/2/A/0396/02/01A, Initial Issue

Nuclear Station Modification (NSM) CN-20396/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the removal of snubbers deleted from Math Model NM-208. This math model includes supports on the Nuclear Sampling (NM) system. These supports will be deleted from the system.

There are <u>no</u> system isolations required to implement this procedure. The only concern is the seismic qualification of the affected system's piping during implementation of this procedure. Math Model NM-208 has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined that this work may be done while the affected system(s) are operable, provided all the support deletions for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified under any combination of snubbers removed. However, if the support deletions cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated that they could justify operable status of any configuration of deleted supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Post Modification Testing is not required for this modification.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/2/A/0396/02/02A, Initial Issue

Nuclear Station Modification (NSM) CN-20396/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the removal of snubbers deleted from Math Model NM-209. This math model includes supports on the Nuclear Sampling (NM) system. These supports will be deleted from the system.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected system's piping during implementation of this procedure. Math Model NM-209 has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined that this work may be done while the affected system(s) are operable, provided all the support deletions for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified under any combination of snubbers removed. However, if the support deletions cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated that they could justify operable status of any configuration of deleted supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Post Modification Testing is not required for this modification.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/2/A/0396/02/06A, Initial Issue

Nuclear Station Modification (NSM) CN-20396/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the removal of snubbers deleted from Math Model NM-205. This math model includes supports in the Nuclear Sampling (NM) system. These supports will either be deleted from the system or revised to a different configuration.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected system's piping during implementation of this procedure. Math Model NM-205 has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined that this work may be done while the affected system(s) are operable, provided all the support deletions for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified under any combination of snubbers removed, provided the modifications to the existing supports have been completed. This procedure is written to complete all modifications before deleting any supports. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated that they could justify operable status of any configuration of deleted supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Post Modification Testing is not required for this modification.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/2/A/0396/02/08A, Initial Issue

Nuclear Station Modification (NSM) -20396/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the removal of snubbers deleted from Math Model NM-207. This math model includes supports in the Nuclear Sampling (NM) system. These supports will either be deleted from the system or revised to a different configuration.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected system's piping during implementation of this procedure. Math Model NM-207 has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined that this work may be done while the affected system(s) are operable, provided all the support deletions for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design

Engineering has performed analyses that indicate the affected piping could be qualified under any combination of snubbers removed, provided the modifications to the existing supports have been completed. This procedure is written to complete all modifications before deleting any supports. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated that they could justify operable status of any configuration of deleted supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Post Modification Testing is not required for this modification.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/2/A/0396/02/16A, Initial Issue

Nuclear Station Modification (NSM) CN-20396/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the removal of snubbers deleted from Math Model BB-201. This math model includes supports in the Steam Generator Blowdown (BB) system. These supports will either be deleted from the system or revised to a different configuration.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected systems piping during implementation of this procedure. Math Model BB-201 has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined that this work may be done while the affected system(s) are operable, provided all the support deletions for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified under any combination of snubbers removed, provided the modifications to the existing supports have been completed. This procedure is written to complete all modifications before deleting any supports. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated that they could justify operable status of any configuration of deleted supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Post Modification Testing is not required for this modification.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/2/A/0396/02/17A, Initial Issue

Nuclear Station Modification (NSM) CN-20396/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the removal of snubbers deleted from Math Model BB-202. This math model includes supports in the Steam Generator Blowdown (BB) system. These supports will either be deleted from the system or revised to a different configuration.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected system's piping during implementation of this procedure. Math Model BB-202 has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined that this work may be done while the affected system(s) are operable, provided all the support deletions for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified under any combination of snubbers removed, provided the modifications to the existing supports have been completed. This procedure is written to complete all modifications before deleting any supports. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated that they could justify operable status of any configuration of deleted supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Post Modification Testing is not required for this modification.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/2/A/0396/02/18A, Initial Issue

Nuclear Station Modification (NSM) CN-20396/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the removal of snubbers deleted from Math Model BB-204. This math model includes supports in the Steam Generator Blow-

down (BB) system. These supports will either be deleted from the system or revised to a different configuration.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected system's piping during implementation of this procedure. Math Model BB-204 has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined that this work may be done while the affected system(s) are operable, provided all the support deletions for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified under any combination of snubbers removed, provided the modifications to the existing supports have been completed. This procedure is written to complete all modifications before deleting any supports. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated that they could justify operable status of any configuration of deleted supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Post Modification Testing is not required for this modification.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/2/A/0396/02/20A, Initial Issue

Nuclear Station Modification (NSM) CN-20396/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the removal of snubbers deleted from Math Model CFA. This math model includes supports in the Main Feedwater (CF) system. These supports will either be deleted from the system or revised to a different configuration.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected system's piping during implementation of this procedure. Math Model CFA has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed, because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined that this work may be done while the affected system(s) are operable, provided all the support

deletions for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified under any combination of snubbers removed, provided the modifications to the existing supports have been completed. This procedure is written to complete all modifications before deleting any supports. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated that they could justify operable status of any configuration of deleted supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported. This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Post Modification Testing is not required for this modification.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/2/A/0396/02/21A, Initial Issue

Nuclear Station Modification (NSM) CN-20396/02 modifies various piping system analyses with the objective of reducing the number of mechanical snubbers. This procedure provides guidance for the removal of snubbers deleted from Math Model BB-203. This math model includes supports in the Steam Generator Blowdown (BB) system. These supports will either be deleted from the system or revised to a different configuration.

There are no system isolations required to implement this procedure. The only concern is the seismic qualification of the affected system's piping during implementation of this procedure. Math Model BB-203 has been qualified for the present support/restraint (S/R) configuration. It has also been qualified for the support/restraint configuration which will be in place after this procedure has been implemented. However, the interim configuration (with some deleted snubbers removed and some still in place) has not been analyzed, because the many possible combinations of S/R configurations would require numerous analyses. For this reason, Design has determined that this work may be done while the affected system(s) are operable, provided all the support deletions for the entire math model are completed within the 72 hours allowed by the technical specification for snubbers. In order to avoid exceeding the 72 hour limit and declaring the affected system(s) inoperable, Design Engineering has performed analyses that indicate the affected piping could be qualified under any combination of snubbers removed, provided the modifications to the existing supports have been completed. This procedure is written to complete all modifications before deleting any supports. However, if the support modifications cannot be completed in the 72 hour time limit, Design Engineering will be contacted to perform an analysis of the affected piping to determine operability. Design Engineering has stated that they could justify operable status of any configuration of deleted supports, provided the affected piping is adequately supported. Station maintenance procedures provide guidance to ensure the piping is adequately supported.

This procedure will provide the necessary controls to ensure compliance with all technical specification requirements.

Post Modification Testing is <u>not</u> required for this modification.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/2/B/0011/00/01A, Initial Issue

Nuclear Station Modification (NSM) CN-20011, Rev. 0 will modify the Ice Condenser piping associated with the floor cooling units on the Ice Condenser Refrigeration (NF) System. This modification will be implemented in parts. TN/2/B/0011/00/01A provides guidance for modifying piping in the lower elevations of the NF system.

The primary concern for the implementation of this modification is melting the ice in the Ice Condenser. With this consideration in mind, this modification will be implemented in parts to minimize the time that the NF system will be out of service. TN/2/B/0011/00/01A involves work activities associated with the lower elevation tie-ins and modifications of the NF system. This work will be performed during the unit 2 outage while the Maintenance Ice Condenser crew is on their scheduled off time. This will enable the NF floor cooling units to be removed from service while minimizing the possibility of ice melting.

Technical Specification 3/4.6.5 discusses the Ice Condenser. This Technical Specification is only required for Modes 1,2,3, and 4 and requires that the temperature be maintained below 27 degrees. The modification will be performed in modes 5, 6, or No Mode. Implementation of this modification will not cause the Ice Bed Temperature Monitoring System to be inoperable; therefore, Operations will still be capable of monitoring the temperature of the Ice Condenser. In addition, this modification will not affect the Ice Bed Doors associated with the NF System.

Section 6.7 of the FSAR discusses the Ice Condenser design. NSM CN-20011 will reroute the Ice Condenser floor cooling supply line from the AHU return line directly to the lower portion of the Ice Condenser compartments. This will enable the NF System to provide cooler glycol to the floor cooling units and will decrease the temperature differential between the lower Ice Condenser and the ice bed. Implementation of this modification does not affect the NF System in a manner which requires a change to the FSAR.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

TN/2/B/0011/00/02A, Initial Issue

Nuclear Station Modification (NSM) CN-20011, Rev. 0 will modify the Ice Condenser piping associated with the floor cooling units on the Ice Condenser Refrigeration (NF) System. This modification will be implemented in parts.

TN/2/B/0011/00/02A provides guidance for modifying piping in the upper elevations of the NF system.

The primary concern for the implementation of this modification is melting the ice in the Ice Condenser. With this consideration in mind, this modification will be implemented in parts to minimize the time that the NF system will be out of service. TN/2/B/0011/00/02A involves work activities associated with the upper elevation tie-ins and modifications of the NF system. This work will be performed during the unit 2 outage while the Maintenance Ice Condenser crew is on their scheduled off time. This will enable the NF system to be removed from service while minimizing the possibility of ice melting.

Technical Specification 3/4.6.5 discusses the Ice Condenser. This Technical Specification is only required for Modes 1,2,3, and 4 and requires that the temperature be maintained below 27 degrees. The modification will be performed in modes 5, 6, or No Mode. Implementation of this modification will not cause the Ice Bed Temperature Monitoring System to be inoperable; therefore, Operations will still be capable of monitoring the temperature of the Ice Condenser. In addition, this modification will not affect the Ice Bed Doors associated with the NF System.

Section 6.7 of the FSAR discusses the Ice Condenser design. NSM CN-20011 will reroute the Ice Condenser floor cooling supply line from the AHU return line directly to the lower portion of the Ice Condenser compartments. This will enable the NF System to provide cooler glycol to the floor cooling units and will decrease the temperature differential between the lower Ice Condenser and the ice bed. Implementation of this modification does not affect the NF System in a manner which requires a change to the FSAR.

Based on this discussion, there are no unreviewed safety questions associated with the implementation of this procedure.

PT/1/A/4600/11, Retype #2, Changes 0 to 6 Incorporated

PT/1/A/4600/11, Neutron Noise Data Acquisition, is not described or mentioned in either Tech. Specs. or the FSAR. The purpose of the procedure is to enable the General Office Neutron Noise Analysis team to safely acquire data from plant instrumentation.

The procedure addresses attaching (and disconnecting) leads to the Operator Aid Computer (OAC) test patch panel, the Incore Instrumentation (ENA) system OAC buffer amps and the Nuclear Instrumentation System (NIS) drawers' test points. The OAC and ENA test points are not safety related and do not have any affect on plant safety or plant parameters. The NIS test points used are buffered test points and will not have any affect on the plant. The procedure addresses the possibility of personnel error in connecting to the wrong test points. The Operator At The Controls (OATC) is warned abou the connection, and caution statements are in the procedure to stop if any alarms occur in the control room.

This procedure does not adversely affect any plant systems or components important to safety. No unreviewed safety question is created.

PT/2/A/4450/05B, Retype, Changes 0 to 12 Incorporated

In Step 6.11, notification of Radiation Protection (RP) upon completion of test was deleted, and step 7.6 was added to replace it. Step 7.6 notifies RP supervision of the possibility of airborne contamination every time the test is performed. "TEST" selector switches were deleted from all sections and verified to be in the correct initial position in section 8.0. When applicable, jumpers are placed to perform the function of the deleted "TEST" selector switches. Deletion of the "TEST" switches allows the Hydrogen Skimmer Fan (HSF) and the Air Return Fan (ARF) to be run at the same time. Section 12.1 was written to verify parameters associated with both the HSF and ARF, and sections 12.4 and 12.5 were included for retest of the fans individually. A jumper with a switch is installed when the same terminals in the Unit 2 Solid State Protection System Train A (2SSPSA) are used in more than one section of the procedure (instead of installing/removing temporary jumpers), and the position of the switch is changed to initiate the appropriate actions. Verification of the "As Found" and "As Left" position of the switches, fans, and dampers is verified once before beginning the procedure and once at the end of the procedure. Each section addresses the correct initial position of the switches, dampers, and fans when applicable. The breaker associated with the HSF is cycled at the end of the procedure to ensure that the trip latch lever has been fully reset per recommendation of the manufacturer.

Required initial test position of the switches, fans and dampers are addressed in each section of the procedure. Problems were encountered with the breaker for HSF-2A tripping during September 1989. If HSF-2B is operated during the performance of this procedure, the procedure requires that the breaker for HSF-2B be cycled to ensure that the trip latch lever has been fully reset per recommendation of the manufacturer. After the breaker is cycled, section 12.6 verifies that power has been returned to HSF-2B by the completion of Enclosure 13.1.

Section 12.1 of this procedure verifies the auto-start of HSF-2B and ARF-2B after an Sp test signal, the 15 minute run time for each fan, the running speed and current for each fan and the Containment Pressure Control Circuitry (CPCC) permissives to start and stop ARF-2B. Two jumpers are placed in 2EATC2, one to open the bypass dampers when ARF-2B starts and one to simulate 2VX2B open to allow HSF-2B to start. Power is removed from ARF-D-4 to prevent an inadvertent opening of the ice condenser doors during the operation of ARF-2B. 2NSPT5240 is placed in the test position, and the test pot is adjusted inside 2CPCC2 to simulate a > 0.4 psid containment pressure, which gives one start permissive for ARF-2B. A jumper in 2SSPSB initiates the Sp signal (timer 2VXTD2 starts), which gives the start signal to both fans after the timer expires. After both fans start, the required alignment of associated dampers is verified, and then the speed and current for each fan is measured. In the event of a LOCA during the operation of the fans in section 12.1, ARF-D-4 would not open and violate the 9 minute time delay for operation of ARF-2B as assumed in the Design Basis Accident (DBA), but 2VX2B would open after the 9 minute delay so that HSF-2B would operate as assumed in the DBA. The Sp test signal is removed before HSF-2B and ARF-2B are shut down to verify fan operation after the Sp signal is removed (as designed). After the 15 minute run time is met, 2NSPT5240 is returned to normal, ARF-2B is verified to stop then the selector switch for HSF-2B is placed in the "Off" position, and

HSF-2B is verified to stop. Applicable jumpers are then removed, and power is returned to ARF-D-4. Section 12.6 verifies that power has been returned to ARF-D-4 by the completion of Enclosure 13.1. Except for the closing of 2ARF-D-4 and the simulation of an open 2VX2B, the above alignment operates HSF-2B and ARF-2B as designed using the proper safety circuitry.

Section 12.2 of this procedure verifies the auto-open of the Air Return Fan discharge damper (ARF-D-4) on an Sp test signal, after the required time delay and the CPCC permissives to prevent/enable the opening of ARF-D-4. After verifying that the CPCC permissive is not initiated, a jumper is placed in 2EATC2 to simulate an Sp signal for ARF-D-4. Then, ARF-D-4 is verified to be closed. The jumper in 2EATC2 is removed, then 2NSPT5250 is placed in the test position, and the test pot is adjusted inside 2CPCC2 to simulate a ≥ 0.4 psid containment pressure, which gives one open permissive for ARF-D-4. A jumper is placed in 2SSPSB to initiate the Sp signal (timer 2VXTD41 starts), which gives the open signal to ARF-D-4 after the timer expires. Since there is a possibility that 2VX2B could open if the Sp signal is left in for 9 minutes, a "CAUTION" statement was added to ensure that the Sp test signal is removed before 2VX2B is given the signal to open. The procedure requires that 2ARF-D-4 be closed immediately after it opens in order to reduce the possibility of Ice Condenser bypass leakage in the event of a LOCA during the test. ARF-2B and HSF-2B are blocked from starting by placing their selector switches in the "OFF" position, to prevent bypass leakage in the event of a LOCA and to ensure that the Ice Condenser doors are not opened during the test. The above alignment opens ARF-D-4 as designed using the proper safety circuitry.

Section 12.3 of this procedure verifies the auto-open of the Hydrogen Skimmer Fan (HSF-2B) suction valve (2VX2B) on an Sp test signal, followed by the start of HSF-2B after the required time delay. The design of the system requires that 2VX2B open or begin to open before HSF-2B will start. There are two 9 minute timers associated with the start of HSF-2B on an Sp signal. One is 2VXTD2, which is the timer that gives a start permissive signal to HSF-2B and ARF-2B, and one is 2VXTD21, which gives the open signal to 2VX2B. This section verifies the correct time for 2VXTD21 (section 12.1 or 12.4 for 2VXTD2). A jumper in 2SSPSB initiates an Sp signal for HSF-2B (timer 2VXTD2 starts), and then at least three minutes later another jumper in 2SSPSB initiates an Sp signal for 2VX2B (timer 2VXTD21 starts). This alignment ensures that the valve opening (after 2VXTD21 expires) is what starts HSF-2B. The procedure requires that HSF-2B be shut down immediately after it starts to limit time that HSF-2B operates with 2VX2B open. A CAUTION statement is included to ensure that HSF-2B is not operated more than 5 minutes with 2VX2B open. ARF-2B is blocked from operation by placing the selector switch in the "OFF" position, to ensure that it will not start in the event of a LOCA during this section and violate the 9 minute time delay. The above alignment operates HSF-2B and 2VX2B as designed using the safety circuitry.

The performance of sections 12.2 and 12.3 of this procedure puts the Containment Air Return and Hydrogen Skimmer System (VX) system in a configuration which deviates from the assumed initial conditions for a Designed Basis Accident (DBA), because the deck leakage area is increased by opening ARF-D-4 and air is moved from Lower to Upper containment by running HSF-2B with 2VX2B open. In both of these alignments, a condition exists for Ice Condenser bypass leakage during a LOCA. The open damper (ARF-D-4) is

addressed in PIR 2-C87-0061, and the conclusion of the PIR was that the effect of the additional steam bypass area on initial compression peak pressure was small. Both sections are required by Tech. Spec. 4.6.5.6 to be performed. Tech. Spec. 3.6.5.5 allows an equipment hatch to be open for 1 hour which also deviates (in the same manner as above) from the initial conditions for a DBA. Caution statements and notes are included in each section to minimize the time that the system is in an alignment as mentioned above. Since the test duration for both sections 12.2 and 12.3 is on the order of a few minutes, not hours, the margin of safety as defined in Tech. Spec. will not be reduced.

Section 12.4 of this procedure verifies the auto-start of HSF-2B on an Sp test signal (after the required time delay), the 15 minute run time, and the fan running speed and currents with 2VX2B closed. The HSF-2B is operated in the same alignment as in section 12.1, and ARF-2B is blocked from operating by placing the selector switch in the "Off" position and by not initiating the CPCC permissive. This section is included for retest purposes, but may be performed instead of section 12.1 if desired.

Section 12.5 of this procedure verifies the auto-start of the Air Return Fan (ARF-2B) after an Sp test signal, CPCC permissives to start and stop the fan, the 15 minute run time, and the fan running speed and current with the discharge damper (ARF-D-4) closed. ARF-2B is operated in the same alignment as in section 12.1, and HSF-2B is blocked from operating by placing the selector switch in the "Off" position and by not opening 2VX2B. This section is also included for retest purposes, but may be performed instead of section 12.1 if desired.

The procedure requires that Containment Air Return Fan 2B (ARF-2B) and Hydrogen Skimmer Fan 2B (HSF-2B) be declared inoperable during the performance of this test. ARF-2A and HSF-2A will remain operable for the duration of the test as required by Tech. Spec. The system is not placed in any unusual alignments. Placement/removal of jumpers, opening/closing of breakers and returning switches, fans and dampers to the required positions are independently verified within section 12.0 or 13.0 of the procedure. Power is verified to be returned to equipment that was deenergized by the procedure on Enclosure 13.1. For these reasons and the ones stated above, an unreviewed safety question does not exist.

PT/2/A/4400/06D, Initial Issue

The test is performed to determine the heat removal capability of the Component Cooling (KC) Heat Exchanger and does not affect its operability during the test. KC Train 2B and the Nuclear Service Water (RN) system will both be aligned for normal operation. If KC and RN temperatures are not stable, 2RN351 will be throttled by use of a regulated air supply in an effort to stabilize temperatures.

Both KC and RN will remain operable during the test. If an air supply is connected to 2RN351 to throttle flow, the valve will still be capable of failing open upon receipt of a S_s signal. If an air supply is connected to 2RN351, the valve will fail open while connecting the air supply, which could cause the KC system to cooldown. This, in turn, could cause a cooldown of the Residual Heat Removal System, which would affect reactivity. If Operations

desires, 2RN347B can be closed to prevent a cooldown. In either case, the procedure has precautions to ensure temperature is closely monitored so that appropriate action may be taken. Steps are also included in the procedure to ensure that there is an adequate safety margin in the event of cooldown.

No unreviewed safety questions are created.

PT/2/A/4250/03C Retype, Changes 0 to 35 Incorporated

The intent of this procedure is to ensure the operational readiness of Turbine-Driven Auxiliary Feedwater (CA) Pump #2, as well as to verify that the Valve Inservice Test (IWV) requirements for several critical check valves are satisfied.

The addition to this procedure of Backflow Testing requirements on the miniflow check valves for CA Pumps 2A and 2B ensures that the probability of malfunction of these check valves will decrease. Likewise, changes included in this retype will ensure increased availability of CA Pump #2, which will, in turn, lessen the consequence of an accident. These changes include the addition of compensatory measures to this procedure, which will allow CA Pump #2 to be considered "Available" during the performance of this test. Likewise, the alignment of critical CA valves within the procedure will minimize the amount of time that CA Pump #2 is in recirculation alignment to the Upper Surge Tank and unable to feed the Steam Generators without requiring Operator action.

CA Pump #2 will be run under normal conditions in recirculation to the Upper Surge Tank. No part of this test requires the pump to be run outside of its normal operating parameters.

Several Limits and Precautions have been added as a result of this retype to ensure that bearing temperatures do not exceed a specified limit during this test, and that the time limit of miniflow operation is observed. These additional precautions will reduce the probability of safety equipment malfunction. Changes have also been included to avoid the system alignment which is conducive to water hammer in the CA discharge piping.

A Limit and Precaution in this procedure has Operations verify that CA Pump #2 can be removed from service for the duration of the test. This will ensure that the two remaining CA Pumps are in service and able to perform their safety function before this test is allowed to proceed. Also, the addition of compensatory measures as described above will ensure that CA Pump #2 is able to perform its safety function with minimal operator action.

The remaining changes included in this retype that are not discussed above are administrative in nature. Therefore, no unreviewed safety questions exist as a result of this procedure retype.

PT/1/A/4250/03C Retype, Changes 0 to 49 Incorporated

The intent of this procedure is to ensure the operational readiness of Turbine-Driven Auxiliary Feedwater (CA) Pump #1, as well as to verify that the

Valve Inservice Test (IWV) requirements for several critical check valves are satisfied.

Several changes included in this retype will ensure increased availability of CA Pump #1, which will, in turn, lessen the consequence of an accident. These changes included the addition of compensatory measures to this procedure, which will allow CA Pump #1 to be considered "Available" during the performance of this test. Likewise, the alignment of critical CA valves within the procedure will minimize the amount of time that CA Pump #1 is in recirculation alignment to the Upper Surge Tank and unable to feed the Steam Generators without requiring Operator action.

CA Pump #1 will be run under normal conditions in recirculation to the Upper Surge Tank. No part of this test requires the pump to be run outside of its normal operating parameters.

Several Limits and Precautions have been added as a result of this retype to ensure that bearing temperatures do not exceed a specified limit during this test, and that the time limit of miniflow operation is observed. These additional precautions will reduce the probability of safety equipment malfunction. Changes have also been included to avoid the system alignment which is conducive to water hammer in the CA discharge piping.

A Limit and Precaution in this procedure has Operations verify that CA Pump #1 can be removed from service for the duration of the test. This will ensure that the two remaining CA Pumps are in service and able to perform their safety function before this test is allowed to proceed. Also, the addition of compensatory measures as described above will ensure that CA Pump #1 is able to perform its safety function with minimal operator action.

The remaining changes included in this retype that are not discussed above are administrative in nature. Therefore, no unreviewed safety questions exist as a result of this procedure retype.

MP/0/A/7150/68 Retype #4, Changes 0 to 1 Incorporated

The changes that have been made to this procedure do not affect its technical content. The changes made include the following:

- Added FSAR Section 6.7 to section 2.0.
- Added additional safety considerations to section 4.3.
- Removed the Shift Supervisor sign off of Section 6.0, as there is a required sign off on the work request.
- Added the ice basket repair parts master sequence number to Section 7.0.
- On the Data Sheet (Enclosure 13.1), all entries of time have been deleted, except for steps 6.1 and 6.2, where the procedure is being verified, and in the Crew Changes section. Experience with this procedure to date has shown that there is no value in having the

technicians record the time for every step that is done. This information has never been used, and it is slowing the work to a small degree.

This procedure has been compared with Tech. Specs., the FSAR, design documents, and station procedures to ensure that the actions it controls will maintain the ice baskets in their as-built/as-designed condition. Therefore, no Unreviewed Safety Question exists.

MP/0/A/7150/67 Retype #11, Changes 0 to 7 Incorporated

This procedure is for performing latching and unlatching operations on the Control Rod Drive Mechanism (CRDM) drive shafts. Instruction manuals CNM-1201.00-0030 and CNM-1201.13-038 provided technical information for the development of this procedure. This evaluation is for changes made following the Unit 1 End of Cycle 5 outage. The Catawba FSAR and Technical Specifications have been reviewed and are not affected by this revision. The probability of an accident or a malfuntion of equipment previously addressed will not be increased nor will any unreviewed safety questions be involved.

This evaluation includes the following changes.

* Step 2.1 — Added reference CNM 1201.00-0030.

* Step 11.6 — Added word "stop" after track for clarification.

* Step 11.14.28 — Removed I.V. from this step.

* Step 11.14.35 — Added this step to perform visual verification of drive rod locking screw position independently of step 11.14.2.

* Revised Enclosure 13.1 Data Sheets for the above changes.

MP/2/A/7150/42 Retype #5, Changes 0 to 4 Incorporated

This procedure is for performing removal and replacement activities required on the Reactor Vessel Head during refueling outages. This evaluation is for changes made during the procedure review following the Unit 1 End of Cycle 4 outage. These changes were required to clarify steps for better understanding and to add manufacture's information on new equipment. See procedure changes below.

* Step 6.6 — Deleted this step from prerequisites. The sequence is covered under step 11.3.4.

* Step 10.3 — *Caution* must change to "should" for temporary installation of covers and seals.

* Step 11.2 — Nil Ductility Transition Temperature (NDTT) changed to Head Flange Temperature requirements for detension/tension sequence as specified in Operations reference procedure.

* Step 11.3.8 — Deleted this ste,). Head stand lead shielding is no longer required due to head stand modification.

* Step 11.4.2 — Changed NDTT to temperature specified in step 11.2.

* Step 11.4.9 — Changed spherical washers to top of the closure stud for clarification.

* Step 11.4.22.1 — Added compensating weight settings.

* Step 11.4.24 — Added cleaning requirements for stud hole plug installation.

* Step 11.4.25 & 11.4.26 — Deleted steps. These steps are covered under guide pin installation procedure.

* Step 11.4.26 — Reworded to add cleaning requirements for alignment pin stud hole locations.

* Step 11.4.27 — Added reference to MP/0/A/7150/63, Guide Stud Removal and Restoration procedure.

* Step 11.4.34 — Added note on match markings.

* Step 11.4.38 — Renumbered to 11.37.4.

* Step 11.4.38 — Added new step for installing head ladder.

* Step 11.4.50 — Added note #4.

* Step 11.5.5 — Added 7 steps for installation of new design o-ring clips per manufacturer's instructions.

* Step 11.5.23 — Added Caution #1.

* Step 11.5.30 — Changed NDTT to temperature specified in step 11.2.

* Step 11.5.61.6 — Deleted. This step was not required. Renumbered the following steps.

* Revised Data Sheets per above changes.

This procedure will be used to maintain the Reactor Vessel head in original design requirements and specifications. The Catawba FSAR and Technical Specifications have been reviewed and are not affected by this procedure change. The probability of an accident or a malfunction previously addressed will not be increased nor will any unreviewed safety questions be involved.