VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

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Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNIT NOS. 1 AND 2
SUMMARY OF FACILITY CHANGES, TESTS AND EXPERIMENTS

Pursuant to 10 CFR 50.59(b)(2), enclosed is a summary description of facility changes, tests, and experiments, including a summary of the safety evaluations, that were conducted at North Anna Power Station during 1993.

If you have any questions, please contact us.

Very truly yours,

M L Bowling

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Enclosure

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1993 50.59 SAFETY EVALUATIONS REPORTABLE TO THE NRC

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DCP 89-048

NON-REGENERATIVE HEAT EXCHANGER
PIPING MODIFICATION
NORTH ANNA UNIT 2

DESCRIPTION

The Chemical and Volume Control System (CVCS) non-regenerative heat exchanger (NRHX) (2-CH-E-2) is 12-pass on the CVCS side and two-pass on component cooling water side. The heat exchanger is installed vertically and the bottom head is divided into seven zones with a 3/4 inch drain line from each zone. Each drain line is connected to a common drain header. There is one drain valve 2-CH-98 at the discharge of this header. This arrangement allows bypass flow between zones which negatively impacts heat exchanger performance. The consequence of not preventing bypass flow between the various zones is to continue to have degraded thermal performance in the heat exchanger.

SUMMARY OF SAFETY ANALYSIS (89-SE-MOD-021)

This design change did not create an unreviewed safety question as defined in 10CFR50.59. A failure of the pressure boundary was considered and the probability and consequences of this accident will not be increased. All secondary break analysis remain fully bounding.

a) The implementation of this modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report.

The NRHX drain piping function solely as a portion of the pressure boundary of the chemical and volume control system and this modification does not alter that function. The addition of another drain valve in series actually decreases the probability of a loss-of-pressure boundary.

b) The implementation of this modification did not create a possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report. The hardware in this modification is of a type used elsewhere in the Station and the modification does not affect the design, function, or operating conditions of the Chemical and Volume Control System. The new drain piping is seismically installed and analyzed.

c) The implementation of this modification did not reduce the margin of safety as defined in the basis of any technical specification. The NRHX is not included in the Technical Specifications. The control, function and operating conditions of the Chemical and Volume Control System was not affected.

This DCP is an old style DCP where the safety evaluation was covered in the Engineering Review and Safety Evaluation (ER&SA). Therefore, a safety evaluation number does not exist.

DCP 93-218

MODIFICATION OF EDG GAUGE PANEL FOR
NEW ENGINE START COUNTER
(SE #90-SE-MOD-102)

DESCRIPTION

The Emergency Diesel Generator (EDG) Engine Start Counter failed on the 1H EDG and is no longer made. The suitable replacement required that the gauge panel be modified to accommodate the new start counter. The DCP provided the necessary direction to modify the panel.

SUMMARY OF SAFETY ANALYSIS

The original safety evaluation for the first start counter replacement was reviewed and determined to be applicable. The safety evaluation concluded that an unreviewed safety question did not exist.

DCP 93-219

MODIFICATION OF EDG GAUGE PANEL FOR
NEW ENGINE START COUNTER
(SE #90-SE-MOD-102)

DESCRIPTION

The Emergency Diesel Generator (EDG) Engine Start Counter failed on the 2J EDG and is no longer made. The suitable replacement required that the gauge panel be modified to accommodate the new start counter. The DCP provided the necessary direction to modify the panel.

SUMMARY OF SAFETY ANALYSIS

The original safety evaluation for the first start counter replacement was reviewed and determined to be applicable. The safety evaluation concluded that an unreviewed safety question did not exist.

DCP 93-135

REPLACE LONERGAN RELIEF VALVE WITH CONSOLIDATED VALVE NORTH ANNA UNIT 1

DESCRIPTION

The refueling water storage tank cooler outlet header relief valve needed to be replaced. The valve was a Lonergan LCT-11 which is no longer available. The relief valve was replaced with a Consolidated 3990 relief valve.

SUMMARY OF SAFETY ANALYSIS (91-SE-MOD-011)

This design change did not create an unreviewed safety question as defined by 10CFR50.59.

A. The replacement of the relief valve with one from a different manufacturer did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

This design change did not affect the operation of the refueling water storage tank or the cooler, in that the ability to maintain RWST level and temperature is not affected. The replacement valve has the same setpoint as the original and will operate the same.

B. The implementation of this modification did not create a possibility for an accident of a different type than previously evaluated in the Final Safety Analysis Report.

The relief valve replaced by this design change meets all of the design requirements of the original and will function the same.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

The replacement valve meets all of the design requirements of the original and will operate in the same manner by lifting when an overpressure condition exists. The margin of safety is not affected.

DCP 92-338

Improve Turbine Driven Aux Feedwater Pump Lube Oil PI Configuration NAPS - Unit 2

DESCRIPTION

The isolation valves to the Turbine Driven Aux Feedwater Pump Lube Oil Pressure Indicators (2-FW-PI-704A,B,C,D) were replaced. The new valve and tubing configuration to the pressure indicators was installed to facilitate maintenance and decrease the number of possible leakage points to only one tubing connection.

SUMMARY OF SAFETY ANALYSIS (91-SE-MOD-031)

The valves and fittings used meet the design requirements of the Aux Feedwater Pump Lube Oil System. The seismic integrity of the Aux Feedwater Pump and the lube oil piping is maintained. System/Pump function and operation was not changed.

Since this modification did not change the function or operation of the Turbine Driven Aux Feedwater Pump there was no increase to the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the UFSAR. No new accidents were created as a result of this configuration change as the pumps still operate as designed. The margin of safety was not affected. Therefore an unreviewed safety question does not exist.

DCP 92-222

REPLACE TEFLON SEALING COMPONENTS ON
CONTAINMENT AIRLOCKS
NORTH ANNA UNIT 1

DESCRIPTION

Because of industry concern about the use of teflon in nuclear containment systems, Chicago Bridge and Iron (CBI), the original vendor of the Containment Personnel Airlock and Escape Lock, has identified that teflon was originally supplied as a sealing material on the Unit 1 Containment Personnel Airlock and Escape Lock. The mechanical properties of teflon have been documented to begin to degrade when exposed to a radiation environment in excess of 1.5 X 10E4 Rads. The normal design radiation environment stated in the applicable Environmental Zone Description reactor containment zone RC-291B exceeds this threshold. The air equalizing valve seats and stem seals in the Containment Personnel Airlock and Equipment Hatch Emergency Airlock were rebuilt to replace the teflon sealing components with modified EPT, which has a radiation resistance of 1 X 10E8 Rads. The teflon shaft seals in the Containment Personnel Airlock Escape Hatches were replaced with tefzel shaft seals which have a radiation resistance of 2 X 10E8 Rads. The radiation resistance for the replacement seal components exceeds the worst case accident radiation environment or 6.79 X 10E6 as described for environmental zone RC-291B (outside of the crane wall). The replacement seal materials envelope all other parameters described by environmental zone RC-291B.

SUMMARY OF SAFETY ANALYSIS - (91-SE-MOD-045)

This design change in accordance with DCP 92-328 does not create an "unreviewed safety question" as defined in 10 CFR 50.59.

A. The implementation of this modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

Replacement of teflon sealing components in the containment airlocks with material more resistent to the effects of radiation does not introduce a significant leakage path. The potential leakage path represented by the air equalizing valve seals and seats and shaft seal is very small due to close clearances on existing metal parts. A double barrier containment feature exists to maintain a margin of safety against leakage.

B. The implementation of this modification does not create a possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report.

The seal materials used in the containment airlocks are passive components which do not have to move to perform their design function during an accident.

C. The implementation of this modification does not reduce the margin of safety as defined in any Technical Specification.

Replacement of teflon sealing materials with materials more resistent to high radiation provides increased margins of safety against containment leakage under high radiation conditions.

DCP 92-223

REPLACE TEFLON SEALING COMPONENTS ON
UNIT 2 CONTAINMENT AIRLOCKS
(SE #91-SE-MOD-045)

DESCRIPTION

Various sealing components on the Unit 2 containment personnel airlock and escape lock were made of Teflon. The original vendor (CBI) identified that the teflon should be replaced with material more resistant to radiation exposure so that seal integrity could be maintained. The air equalizing valve seats and stem seals were rebuilt with modified EPT and tefzel. The teflon shaft seals were replaced with tefzel shaft seals.

SUMMARY OF SAFETY ANALYSIS

The safety evaluation concluded that an unreviewed safety question does not exist.

DC 90-15-1
PREPARATION FOR STEAM GENERATOR REPAIR
NORTH ANNA - UNIT 1

DESCRIPTION

There were five (5) steam generator replacement prerequisite activities which were performed during the mid-cycle outage which was prior to the actual steam generator replacement outage. These activities provided modifications to facilitate the rigging and transport of the new and old steam generators. Modifications included concrete cutting, relocation of electrical and tubing interferences, installation of anchor bolts for the auxiliary crane and runway beams, replacement of the equipment hatch floor and modification of the equipment hatch platform outside containment.

1. Concrete Cutting

   a. Sections of the bioshield walls for the steam generators were cut above the operating deck and re-secured with through bolted splice plates. This modification allowed subsequent removal during the replacement outage.

   b. An overhead section 1' X 8' of the polar crane wall at the equipment hatch was removed to allow passage of the steam generator transition cone.

   c. A 5' X 8' X 10" section of the operating deck located in front of the equipment hatch was removed to allow alignment of the transition cone with the hatch barrel. A removable steel floor was installed in place of the removed concrete section.

2. Relocation of Electrical and Tubing Interferences

   Electrical conduits, receptacles and instrument tubing supports were permanently relocated off the removable sections of the steam generator biological shield walls. Additionally, a lighting fixture on the removed section of the polar crane was relocated.

3. Anchor Bolt Installation

   During steam generator replacement runway beams for the transport system and an auxiliary crane were installed. This design change located the baseplates for the runway beam and auxiliary crane, drilled bolt holes in the operating deck and installed the inserts.

4. Equipment Hatch Barrel Floor

The existing concrete floor in the equipment hatch barrel was removed to provide clearance for the transition cone portion of the steam generator lower assembly for transport through the equipment hatch barrel. The existing floor was permanently replaced with a steel floor that could be removed during the replacement outage. A temporary restraint for the new steel floor was installed to ensure that there is no potential damage to the equipment hatch under seismic loading during plant operation. After steam generator replacement, the steel floor was welded to the barrel to preclude movement under seismic loading.

5. Equipment Hatch Platform

The equipment hatch platform outside the containment building was modified to accommodate both the steam generator replacement and normal refueling activities.

a. The platform beam directly in front of the equipment hatch was permanently modified to provide clearance for the transition cone portion of the steam generator lower assembly as it moves in or out of the equipment hatch barrel.

b. To facilitate movement of the steam generator lower assemblies, additional footings and structural steel was installed to widen the existing platform, to add a new stair tower, and to extend the existing platform. The existing platform and the platform extension was temporarily configured so as to adequately support the steam generators.

c. After rigging of the steam generator lower assemblies was completed, additional footings and structural steel were added to make the platform extension and the stair tower into permanent, stand-alone structures. The stair tower and the platform extension was disconnected from the existing platform prior to startup from the steam generator replacement outage. The structural steel added to widen the existing platform remained attached to the existing platform.

SUMMARY OF SAFETY ANALYSIS - 91-SE-MOD-053

Modifications implemented by this design change did not create an unreviewed safety question as defined by 10CFR50.59. A discussion of the five (5) activities follows.

1. Concrete Cutting

   The containment structure is designed to sustain, without loss of required integrity, all effects of gross equipment failures up to and including the rupture of the largest pipe in the RC system and any condition resulting from a LOCA. The in-containment structural modifications do not affect the performance or integrity of the containment. The modifications to the steam generator biological shield wall and the section of the operating floor have been analyzed and found structurally acceptable under normal and accident loadings. The substitution of the structural steel platform for the removed section of the operating floor did not affect seismic loads. The permanent enlargement of the opening in the polar crane wall was analyzed and found structurally acceptable under seismic and crane loading.

2. Relocation of Electrical and Tubing Interferences

   Relocation of electrical conduits, receptacles, light or instrument tubing supports meets plant specifications and does not affect any safety function.

3. Anchor Bolt Installation

   The modifications associated with the auxiliary crane and runway beam anchor bolts did not involve cutting rebar and, therefore, a seismic analysis was not necessary.

4. Equipment Hatch Barrel Floor

   Replacement of the equipment hatch barrel floor did not affect the structural integrity of the equipment hatch. Detailed work procedures were developed to ensure that the equipment hatch barrel is not damaged during removal of the existing floor. The equipment hatch barrel was inspected after the work was completed to verify that it that it had not been damaged by the work, or that any incidental damage had been acceptably repaired. Additionally, the new floor has the same load-bearing capacity as the original design.

5. Equipment Hatch Platform

The existing equipment hatch platform and the platform extension have been evaluated for the loads imposed by the removal and installation of the original/new steam generator lower assemblies, as well as other equipment. The structural adequacy of these structures, as temporarily configured for the steam generator replacement outage, is documented in Calculations 20559-C106-01 and 20559-C206-02. The temporary configuration of the stair tower was also analyzed in Calculation 20559-C106-01.

The permanent, stand-alone, configurations of the stair tower and the platform extension were analyzed and found to be structurally adequate in Calculations 21809-C-016 and 21809-C-017, respectively. Although these structures are classified as non-seismic, the analyses also concluded that the platform extension and the stair tower will not fail and impact the existing platform under seismic loading conditions. Consistent with UFSAR Sections 3.3.2 and 3.5.4, if portions of the platform extension, stair tower, or jib crane impacted the existing platform, the missile shield, or the containment during a tornado, a condition no worse than previously evaluated for the design-basis utility pole missile would result.

The existing equipment hatch platform, although not previously classified, is now safety-related. This platform supports the labyrinth portion of the missile shield which protects a portion of the equipment hatch. A structural analysis of the permanent configuration of the existing platform was performed (reference Calculation 02072.4810-S-3). Results of the analysis confirm that all stresses are within applicable UFSAR allowables under seismic loading conditions. For the tornado wind loading condition, preliminarily results indicate that the existing platform is potentially overstressed beyond UFSAR allowables in the columns near their bases and in the concrete piers supporting the columns. However, the analysis preliminarily concludes that the existing platform would remain functional under tornado wind loads and that the labyrinth would continue to provide missile protection of the equipment hatch. A separate 10CFR50.59 safety evaluation and Justification for Continued Operation was prepared by Virginia Power to address the potential overstress condition.

The existing platform and the labyrinth portion of the missile shield are considered to be a single structure and meet the tornado missile criteria in UFSAR Sections 3.3.2 and 3.5.4. Specifically, the missile protection afforded the containment equipment hatch by the labyrinth/existing platform satisfies the UFSAR Section 3.3.2 criterion that the design should assume maximum wind forces and partial vacuum to occur concurrent with a single design basis missile impact.

All of the modifications for this activity were conducted outside the containment building. There were no buried or adjacent safety-related facilities which could be adversely impacted by the work activities. Modifications to the existing platform that require removal of the tornado missile shield were performed when the plant was in Modes 5 or 6. Prior to startup from the steam generator replacement outage (ie., entry into Mode 4), the stair tower and platform extension were disconnected from the existing platform. Therefore, accidents and malfunctions analyzed in the UFSAR are not be affected by this activity.

SAFETY EVALUATION NUMBER 91-SE-MOD-056 Rev 1

DESCRIPTION

Provide ventilation for demin alley sumps during reactor coolant and letdown filter changeouts.

SAFETY EVALUATION SUMMARY

The modification is self-contained, imparts insignificant loads to SR equipment, and does not affect the margins of existing SR systems, structures or components, an unreviewed safety question does not exist.

DC 87-007-1

Fuel Transfer System
North Anna / Unit 1

Description

The fuel transfer system is the equipment required to transport fuel assemblies between the Reactor Refueling Pool and the Spent Fuel Pool. It consists of the transfer tube, the conveyor car and the upenders.

The Conveyor Car is a horizontal structure with large wheels on each side that support the car and allow it to roll on rails within the transfer tube. The car, in turn, supports and moves the fuel assembly container between the two pools. The conveyor car is approximately 35 feet long, with the 15-foot section nearest the Spent Fuel Pool end used for carrying the fuel assembly container. A single drive chain tack welded to the bottom of the cart was used to move the cart back and forth inside the transfer tube. The chain was engaged by sprockets on the containment side which were driven through a roller chain by an air motor located underwater. The conveyor car drive system resembled a rack and pinion arrangement, the chain attached to the bottom of the conveyor car acting as the rack, and the mating sprockets functioning as the pinion.

North Anna Power Station had experienced a variety of breakdowns in the Fuel Transfer System. Breakdowns had occurred in the chain drive system, air motor, proximity switches and emergency pull-out cable. These repeated breakdowns were costly to the Station because they directly impacted the length of refueling outages. In addition, these repeated breakdowns increased radiation exposures to Station maintenance personnel.

A component that had frequently failed was the underwater air motor. Water in the air line had led to internal rusting and had resulted in motor failure. The underwater proximity switches were used to indicate the position of the transfer tube valve as well as sense the final position of the conveyor car. These switches had failed in the past and had come out of adjustment during refueling operations, negating some key interlocks required to prevent operator errors in handling fuel assemblies. The emergency pull cable consisted of a stationary cable between the shear pin located near the transfer tube and a stationary pulley located on the canal wall opposite the transfer tube. The cable was used to return the car to the reactor side if failure of the drive system occurred. On several occasions, the cable had become entangled with the conveyor car and caused delays in the refueling process.

This design change modified the Fuel Transfer System by replacing the underwater air motor and chain drive system with a cable drive system. The air supply line to the reactor containment control panel (which powered the air motor) was cut off and capped. The cable drive system consists of two electric motor driven winch drums located on the operating floor above the reactor transfer canal. Both electric winches are energized to provide mutually opposing torque whenever movement is required and the brakes on both winches are released during carriage travel. The direction of travel is controlled by providing high torque to the motor in the direction of travel and lower counter torque to the opposite winch motor to maintain cable tension. The torque is varied on a wound rotor type motor by varying the resistance in the rotor circuit.

The underwater limit switches used to limit travel of the fuel transfer car were replaced with programmable limit switches that monitor the rotation of the winch. A new limit switch for the transfer tube valve was relocated above water: a spring loaded cable was mounted to the valve stem transmitting relative gate valve position to the limit switch. Two limit switches, one associated with each of the upender winches, located a short distance below the water surface were replaced by identical switches to ensure proper operation.

The emergency pull-out cable assembly was removed. The new cable drive system incorporates a secondary means of cart retrieval via manual handwheels installed on the two electric winches located on the operating floor.

The control panels for the Unit 1 Fuel Transfer System in the Fuel Building and the Unit 1 Reactor Containment were replaced with new panels. The new panels eliminated solenoid operated valves associated with the air drive system and included all new programmable limit switches. Control panel interlocks and operation remained largely unchanged. The air drive system was demolished to the extent possible (limited mainly by eliminating impediments to the new system and radiological conditions).

Summary Of Safety Analysis (91-SE-MOD-060)

This modification was reviewed to determine if an unreviewed safety question as defined in 10CFR50.59 existed. Consequently, no unreviewed safety questions were known to exist as a result of the change. The result of this evaluation can be stated as follows:

1) The implementation of this Design Change did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR. This design change modified the method of moving the fuel transfer car and replaced underwater limit switches with above water limit switches, which are more sensitive and

reliable. These features decrease the probability of a fuel-handling accident inside containment, as described in Section 15.4.7 of the UFSAR.

2) The implementation of this modification did not create a possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR. This Design Change modified the method of moving the fuel transfer car and replaced underwater limit switches with above water limit switches, which are more sensitive and reliable. These features were improvements in the design, and therefore did not create the possibility for an accident different from the type described specifically in section 14.4.1 of the UFSAR.

3) The implementation of this modification did not reduce the margin of safety as defined in the basis of the Technical Specifications. Only the method of conveyor car movement was changed, thus the margin of safety was not affected or reduced in the basis of TS 3/4.9.

DCP 91-125

CONTAINMENT AIR SUPPLY AND PURGE
SYSTEM DUCT SUPPORT MODIFICATIONS
(SE #91-SE-MOD-065)

DESCRIPTION

While conducting walkdowns for the Electrical Distribution System Functional Assessment (EDSFI) it was noticed that HVAC duct supports linked the Containment walls and Rod Drive Control Rooms of the Auxiliary Building together. The buildings are on separate foundations and have a 2 inch rattle space between them. Modifications were made to three different duct supports to permit relative displacement during a seismic event so that damage to supports and/or ductwork could not occur.

SUMMARY OF SAFETY ANALYSIS

The safety evaluation addressed the impact on the Containment Air Supply and Purge System, and evaluated proper seismic loading cases for the modification. The safety evaluation concluded that an unreviewed safety question does not exist.

DCP 91-118

ADDITION OF WELDING RECEPTACLE IN MAIN STEAM VALVE HOUSE NORTH ANNA / UNITS 1 & 2

DESCRIPTION

There were no 480V power receptacles located in the Main Steam Valve House. Extension cords were required to be used to power welding machines in this area. To provide permanent power for the welding machines a 480V outlet was added in each unit MSVH. This modification provided a safer more convenient means of providing power to these welding machines.

SUMMARY OF SAFETY ANALYSIS (Safety Evaluation #91-SE-MOD-068)

The installation of 480V welding receptacles in the MSVH does not constitute an "unreviewed safety question" as defined in 10CFR50.59 because:

A. The implementation of this modification did not increase the probability of occurrence, or the consequences of, an accident or malfunction of equipment important to safety and previously evaluated in the Updated Final Safety Analysis Report.

The seismically mounted conduit and receptacle has no potential to harm any surrounding safety related equipment because seismic mounting is designed to withstand a design basis seismic event.

B. The implementation of this modification did not create a possibility for an accident or a malfunction of a different type than any evaluated previously.

The receptacle added by this modification has no accident scenario postulated which would threaten any other systems performance.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

The seismically mounted welding receptacles that were installed have no interaction with safety related systems.

DCP 91-004

SERVICE WATER 4" CR CHILLER PIPING REROUTE
NORTH ANNA UNIT 2

DESCRIPTION

During investigation of service water (SW) system corrosion problems evidence was found that indicated leakage from the 4" supply and/or return piping to/from the Unit 2 Control Room Chillers.

This design change replaced the 4" supply and return piping to the Unit 2 Control Room Chillers. Because the original piping was encased in concrete, the new piping was routed from existing 24" headers in the Auxiliary Building. The old piping was isolated from the 36" diameter headers and abandoned in place.

The overall philosophy for implementation of this modification was similar to the DCP 89-01-3 project which rerouted the 4" lines on Unit 1. Salient features of the project are listed below:

1. New 4" lines (total of 4) were routed from the Auxiliary Building (AB) in the area of the 24" headers to the Component Cooling (CC) heat exchangers, through the Cable Vault, Air Conditioning Room and into the Chiller Room.
2. New isolation valves were provided at the point of origin (in AB) and the Chiller Room. Four new stainless steel valves were in stock (surplus from DC 89-01-3) for use in the AB. New valves were purchased for and installed in the Chiller Room.
3. Piping materials are Type 316L stainless steel, the current material-of-choice for the SW routing within the station buildings.
4. Piping as sleeved (as with Unit 1) within the Cable Vault and Air Conditioning Room due to presence of sensitive electrical components and flooding considerations.
5. Abandonment of the existing 4" lines required personnel entry into the 36" diameter SW headers under the floor of the Service building to plug the 4" connections.
6. Piping replacement involved 4" piping up to the first convenient break point (i.e. Flange, component, etc.) within the Chiller Room and did not replace components or piping in the Chiller Room other than that necessary for the tie-in.

SAFETY EVALUATION NUMBER 91-SE-MOD-078 Rev 1

DESCRIPTION

In containment structural modifications to provide clearance for passage of S/G lower assemblies.

SAFETY EVALUATION SUMMARY

The in-containment structural modifications do not affect the performance or integrity of the containment. The existing equipment hatch platform would remain functional under tornado wind loads.

SUMMARY OF SAFETY ANALYSIS (92-SE-MOD-003)

The Design Change did not involve an unreviewed safety question, as the goal of this change was to reroute SW pipes to the Unit 2 CR chillers to replace the deteriorated original piping.

a) The probability of occurrence or the consequence of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR was not reduced. Flow rates and flow pattern of the service water system and components remain unchanged under normal and accident conditions. The new piping is safety related and seismically designed. Safety related structures along the new pipe routing have been evaluated under the conditions outlined and are unaffected.

b) The possibility for an accident or malfunction of a different type than any evaluated previously in the UFSAR has not been created. Operability of safety related components (as defined by the ability to perform the intended safety function) remains unaffected. Implementation of the design change does not create the possibility for a different type of accident nor does this change have any impact on the functional capability of safety related components.

c) The margin of safety as defined in the basis for any Technical Specifications has not been reduced. Major construction work was performed during a Unit 2 refueling outage to eliminate any possible affect on the safe operation of the Unit.

DCP-91-021-1
EQ TEMPERATURE MONITORING
NORTH ANNA POWER STATION
UNIT 1

DESCRIPTION

To address the issues of high temperature environments and thermal qualification of EQ equipment, a temperature monitoring system was installed in accordance with NAPS Engineering Work Request (EWR) 89-322 to record ambient temperatures in areas containing EQ equipment inside North Anna Unit 1 & 2 Containment and Main Steam Valve House. Data retrieved by this system enables engineers to more accurately predict the design life of the subject components. Additional locations have been identified inside the Unit 1 & 2 Containment Annulus and in the Auxiliary Building where EQ equipment may be exposed to elevated temperatures. Additional thermocouple placements were necessary to obtain temperature data and characterize the ambient conditions in these new areas of concern.

Thermocouples installed by EWRs 89-322 and 88-328 input temperature data to a data logger (1RC-DAL-101), installed by EWR-88-328. This data logger was installed to monitor temperatures of the reactor coolant piping and the pressurizer surge line in response to NRC Bulletins 88-08 and 88-11 concerning thermal stressed in RC piping. The data from this system was originally required for only one refueling cycle. DCP 90-07-3 extended the installation for one additional fuel cycle.

The purpose of this Design Change was to provide the design for using spare thermocouple channels installed by EWR 88-328 to monitor new locations near EQ equipment in the Unit 1 Containment and Auxiliary Building Safeguards Area.

Additionally, six thermocouples installed by EWR 89-322 which have proved to be located in low temperature areas were relocated. All other existing instrumentation installed by EWR 88-328, EWR 89-322, and DCP 90-07-3 was extended for another fuel cycle.

SUMMARY OF SAFETY ANALYSIS (92-SE-MOD-008)

This design change does not create an unreviewed safety question as defined in 10CFR50.59.

The implementation of this design change does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR. The thermocouples installed are part of a passive, temperature monitoring system that does not interface with any other plant system.

The implementation of this design change does not create a possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR. The modification described by this design change does not affect the design, function, or operating conditions of any other plant system.

The implementation of this design change will not reduce the margin of safety as defined in the technical specifications. Addition of the described thermocouples will not affect the control, function, or operating condition of any plant system.

The thermocouples and associated cabling installed by this design change are small in size and weight and their failure in a seismic event will not jeopardize the equipment that the thermocouples are supported from nor will their failure affect any surrounding safety related equipment.

INSTALLING LEVEL INDICATORS
01-CN-LI-100A/100B1
NORTH ANNA UNIT 1

DESCRIPTION

The existing Emergency Condensate Storage Tank (01-CN-TK-1) level indicator (01-CN-LI-100B-1) located in the control room on vertical board 1-2 had failed. This indicator is an International Instruments, Inc. Lumigraph (bar indicator), model number 9270-11-D-VB, and is no longer manufactured. An acceptable replacement is required.

This design change replaced the existing level indicators with new VMI series 2000 indicators. These indicators are equipped with a single highly visible vertical bar graph comprised of vacuum florescent displays (VFD's) and an accompanying digital readout. The digital readout is located directly beneath the vertical bar graph. The bar graph and the digital readout will indicate the actual level in the tank. Under normal conditions, the bar graph is red from top to bottom and the digital readout will be 100, indicating the tank is full. A single VFD will be a brighter red than the others indicating the alarm setpoint, which will be set by the instrument shop. When the auxiliary feedwater is supplied by the emergency condensate storage tank, the red bar begins darkening at the top and the digital readout lessons as level decreases in the tank. When the level indicated on the bar graph decreases to the alarm setpoint indicated by the brighter VFD, a low alarm red L.E.D. will illuminate on indicator. The low alarm L.E.D. is located on the top right of the indicator.

SUMMARY OF SAFETY ANALYSIS (92-SE-MOD-013)

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

A. The implementation of this modification did not increase the probability of occurrence or consequences of an accident or malfunctions of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

This design change did not affect the operation of the safety-related emergency condensate tank level indicators in the control room or the design basis of the system to perform its intended function. This design change replaced the level indicators with a like-for-like component and the operation of the indicators will remain the same.

B. The implementation of this modification did not create a possibility for an accident or a malfunction of a different type than any previously evaluated in the Final Safety Analysis Report.

The level indicators installed by this design change have no accident scenario postulated which would threaten any other system's performance. There is no interaction between the new level indicators and any other system.

C. The implementat on of this modification did not reduce the margin of afety as defined in the basis of any Technical Speci ication.

The level indicators installed by this design change will not affect the level indication of the condensate storage tank in the control room. This is a one for one level indication replacement. Thus, no impacts on safety margins have been created.

DCP 92-106

LOW HEAD SAFETY INJECTION SYSTEM
PUMP DISCHARGE PIPING VENT ADDITIONS
NORTH ANNA UNIT 1 AND 2

DESCRIPTION

When the Low Head Safety Injection (LHSI) Pumps are started, the relief valves in the pump discharge piping momentarily lift. A Type One Report prepared by Mechanical Engineering entitled "Evaluation of LHSI Pump Relief Valve Discharge" concluded that the cause of the relief valve lifts, subsequent to LHSI pump start, was due to pressure transients resulting from the compression of entrapped air in the system piping. Ultrasonic testing of accessible Unit #1 and Unit #2 safety injection piping was conducted and it was determined that the installation of additional piping vents and large system valve bonnet vents was warranted. It was also determined that the set pressure for Unit #1 LHSI pump discharge relief valves, 1-SI-RV-1845A, B, & C could be increased to 257 psig to reduce the challenges to the relief valves.

In Unit #1, high point vent valves were added to the inlet elbow to 1-SI-MOV-1890D in line 10"-SI-239-153A-Q2 and to line 4"-SI-258-153A-Q2 near valve 1-SI-315. Bonnet vent valves were added to the following nine existing Unit #1 valves: 1-SI-9, 1-SI-26, 1-SI-MOV-1864A & B, 1-SI-MOV-1890C & D, 1-SI-195, 1-SI-197, and 1-SI-199. The set pressure for Unit #1 LHSI pump discharge relief valves 1-SI-RV-1845A, B, & C were increased to 257 psig.

In Unit #2, a high point vent valve was added to the inlet elbow to 2-SI-MOV-2890D in line 10"-SI-623-153A-Q2.

SUMMARY OF SAFETY ANALYSIS (92-SE-MOD-014)

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

A. The implementation of this modification did not increase the possibility of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

This design change did not affect the operation of the safety-related Low Head Safety Injection System or the design basis of the system. The additional vent valves allow proper venting of the system prior to performance of periodic tests or placing the system in operation. The vent valves, piping, and fittings installed meet or exceed the design requirements for the portion of the system for which they are installed. The valve bonnet vent designs were approved by the appropriate valve manufacturer. The new relief valve set pressure does not exceed the design pressure of any LHSI system components presently in the system.

B. The implementation of this modification did not create a possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report.

The modifications made by this design change are mechanical modifications which did not change the operation or design basis of the LHSI system or any other system. No new accident scenarios were postulated for the LHSI system or any other system due to this design change.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

The added vent valves are physically passive components and have no impact on safety margins. The revised set pressure of Unit #1 relief valves 1-SI-RV-1845A, B, & C did not exceed the design pressure of any LHSI system components presently in the system and did not change the system design pressure. Therefore the revision of the relief valve set pressures had no impact on safety margins.

DCP 92-175

REMOVAL OF 3 PIECE CONCRETE BLOCK FROM U2 CONTAINMENT
(SE #92-SE-MOD-28)

DESCRIPTION

Three piece concrete blocks (ID 24, 25, and 26) were permanently removed from the Unit 2 containment in order to decrease time spent during the outage for the "block shuffle". The blocks were used during the outage for laydown space, they had no operational significance.

SUMMARY OF SAFETY ANALYSIS

The safety evaluation concluded there is no unreviewed safety question and the appropriate design basis calculations were updated.

DCP 91-019

WASTE OIL TANK REPLACEMENT, GASOLINE TANK AND GASOLINE ISLAND REMOVAL AND INSTALLATION (SE #92-SE-MOD-031)

DESCRIPTION

Virginia State law mandated that various gasoline and oil tanks be replaced to preclude leakage into soils and underground water aquifers. The DCP replaced the Waste Oil Tank with a new 6000 gallon tank. The DCP also removed the existing 1000 gallon and 3000 gallon tanks and the pump island from their original location. This was replaced with a new 10,000 gallon tank and island closer to the vehicle control area.

SUMMARY OF SAFETY ANALYSIS

The safety evaluation addressed temporary changes to the Security facility and addressed temporary waste oil disposal methods. The safety evaluation concluded that an unreviewed safety question does not exist.

DCP 92-117

**DETERMINATION OF CABLES
IN LIQUID WASTE DISPOSAL SYSTEM
NORTH ANNA / UNIT 1**

DESCRIPTION

Liquid Waste evaporate distillate test tanks '-LW-TK-5A and 5B are used as temporary storage tanks for Component Cooling water. EWR 86-373A documents a UFSAR revision to eliminate operability requirements for the liquid waste evaporators. The evaporate distillate tanks are no longer used in the Liquid Waste System. The conductivity cells associated with the tanks are no longer required and were removed by this modification. The UFSAR had not been updated to reflect the storage of Component Cooling water in these tanks. This UFSAR was revised to reflect storage of Component Cooling Water in these Liquid Waste Tanks.

SUMMARY OF SAFETY ANALYSIS (Safety Evaluation # 92-SE-MOD-033)

The determination of cables does not constitute an "unreviewed safety question" as defined in 10CFR50.59 because:

A. The implementation of this modification did not increase the probability of occurrence, or the consequences of, an accident or malfunction of equipment important to safety and previously evaluated in the Updated Final Safety Analysis Report.

   The equipment and cables being modified are non-safety related.

B. The implementation of this modification did not create a possibility for an accident or a malfunction of a different type than any evaluated previously.

   The system being modified is non-safety related, not required to function during an accident, and not normally used during normal plant operation.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

   This modification does not affect any part of Technical specifications.

DCP 91-018

CONTAINMENT INSTRUMENT AIR RECEIVER REPLACEMENT
NORTH ANNA UNITS 1&2

DESCRIPTION

The end caps on the containment Instrument Air (IA) receiver tanks (1/2-IA-TK-2A&B) were found to be deteriorated due to moisture previously present in the IA system. The root cause of the deterioration was addressed by upgrading the instrument air system under DCP 89-04. The deterioration of the tanks had progressed to the extent that the end caps on all four of the containment IA receivers had degraded below minimum wall thickness.

The design change upgraded the containment IA system by replacing the four existing carbon steel containment IA receivers with new carbon steel tanks containing a corrosion allowance. In addition, the relief valve setpoint for each tank was increased to reflect the new tank design pressure.

SUMMARY OF SAFETY ANALYSIS (92-SE-MOD-036)

The replacement of the deteriorated containment IA receivers did not constitute an unreviewed safety question as defined in 10CFR50.59 since it did not:

A) Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in UFSAR.

The receiver tank replacement was essentially a one-for-one replacement with new tanks having a greater corrosion allowance which did not affect any operations or ability of equipment important to safety to perform their safety functions.

B) Create a possibility for an accident or malfunction of a different type than any evaluated previously in the UFSAR.

The Design Change increased the reliability of the containment IA system by replacing the containment IA receivers with new tanks containing a corrosion allowance. The function of the new receivers did not change.

C) Reduce the margin of safety as defined in the basis of any Technical Specification.

Implementation of the Design Change improved the reliability of the containment IA system. No margin of safety was reduced or impacted for the basis section of the Technical Specifications.

INSTALL REMOTE GETARS TRIP SWITCH IN THE MAIN CONTROL ROOM NORTH ANNA UNIT 1

DESCRIPTION

The plant has experienced abnormal operating transients during the past couple of years. The operating conditions of the plant during the transient conditions is rarely ever recorded unless the Unit actually trips. Thus, no data is available to perform a root cause analysis of why the transient occurred. A General Electric Transient Analysis Recording System (GETARS) trip switch was installed in the control room on the wall below 1MUX-33A and behind the boron recovery (1-EI-CB-13) cabinet. This trip switch allows the Shift Technical Advisor (STA) to trip the GETARS from the control room during a transient condition.

SUMMARY OF SAFETY ANALYSIS (92-SE-MOD-037)

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

A. The implementation of this modification did not increase the probability of occurrence or consequences of an accident or malfunctions of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

This design change conforms to standards and administrative procedures. The GETARS computer is not utilized for a safety related function, but utilized for plant transient recording just prior to and after a unit trip condition. The GETARS and ERF computer system both connect to the validyne multiplexer, which is classified as NSQ. The trip switch was interfaced to a separate digital input card in the validyne multiplexer. Loss of the validyne multiplexer or the GETARS computer system does not increase the probability of an accident from occurring.

B. The implementation of this modification did not create a possibility for an accident or a malfunction of a different type than any previously evaluated in the Final Safety Analysis Report.

The design change was minor and did not affect the operation of the ERF computer system.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

The validyne multiplexer and GETARS computer system are not discussed in the Technical Specifications. Thus, the margin of safety has not been affected.

DCP 92-100

CHARGING PUMP SPEED INCREASER OIL
PRESSURE GAUGE INSTALLATION
NORTH ANNA UNIT 1

DESCRIPTION

The charging pump speed increasers had an excessive amount of oil vapor coming out of the speed increaser breather caps. The speed increaser oil is supplied from the bearing supply oil. The oil pressure for the bearings was being set using a gauge on the inlet of the speed increaser making both oil pressures the same. Per the manufacturer, the bearing oil pressure is 15-20 psig but the speed increaser only requires enough oil to coat the gears. An additional pressure gauge was installed at the outlet of the oil cooler and the original gauge to the speed increaser was changed to one with a lower scale in order to monitor oil to the gear spray.

SUMMARY OF SAFETY ANALYSIS (92-SE-MOD-039)

This design change did not create an unreviewed safety question as defined by 10CFR50.59.

A. The gauge installation did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

The design change did not affect the operation of the safety related charging pumps or the design basis of the charging/safety injection systems to perform their intended functions. The gauges are used to monitor oil pressure to the speed increasers excess oil spray.

B. The implementation of this modification did not create a possibility for an accident of a different type than previously evaluated in the Final Safety Analysis Report.

The gauges added by this design change are for monitoring oil pressure only. There is no interaction between these gauges and any equipment which could create the possibility for an accident of a different type.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

Charging pump operation was not affected as the gauges are used only to set and monitor oil pressure accurately. The margin of safety is not affected.

DCP 92-242

CHARGING PUMP SPEED INCREASER OIL PRESSURE GAUGE INSTALLATION NORTH ANNA UNIT 2

DESCRIPTION

The charging pump speed increasers had an excessive amount of oil vapor coming out of the speed increaser breather caps. The speed increaser oil is supplied from the bearing supply oil. The oil pressure for the bearings was being set using a gauge on the inlet of the speed increaser making both oil pressures the same. Per the manufacturer, the bearing oil pressure is 15-20 psig but the speed increaser only requires enough oil to coat the gears. An additional pressure gauge was installed at the outlet of the oil cooler and the original gauge to the speed increaser was changed to one with a lower scale in order to monitor oil to the gear spray.

SUMMARY OF SAFETY ANALYSIS (92-SE-MOD-039)

This design change did not create an unreviewed safety question as defined by 10CFR50.59.

A. The gauge installation did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

   The design change did not affect the operation of the safety related charging pumps or the design basis of the charging/safety injection systems to perform their intended functions. The gauges are used to monitor oil pressure to the speed increasers excess oil spray.

B. The implementation of this modification did not create a possibility for an accident of a different type than previously evaluated in the Final Safety Analysis Report.

   The gauges added by this design change are for monitoring oil pressure only. There is no interaction between these gauges and any equipment which could create the possibility for an accident of a different type.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

   Charging pump operation was not affected as the gauges are used only to set and monitor oil pressure accurately. The margin of safety is not affected.

DCP 92-123

REPLACEMENT OF MAIN CONTROL ROOM CHILLER SERVICE WATER STRAINER

DESCRIPTION

The strainers which were originally installed in the Service Water supply to the condenser section of the Control room Chillers were basket type strainers with a self-cleaning feature. These strainers were designed to shift sides automatically at a preset differential pressure so that the clean side of the strainer was in service and the dirty side would backflush. Corrosion and 12 years of use had caused the mechanisms in the strainers to hang up periodically.

The strainers were all replaced with stainless steel wye-type strainers. The flush connection from the strainer basket of each of these strainers has an isolation valve and is piped to the Service Water return piping. The equipment which was associated with the self-cleaning feature of the original strainer was removed. When pressure drop across a strainer becomes excessive it has to be manually flushed. A local differential pressure gauge allows the operators to monitor differential pressure across each strainer. Instruction for flushing the strainers have been added to the operator log sheets.

SUMMARY OF SAFETY ANALYSIS (92-SE-MOD-043)

Replacement of the self-cleaning strainers with wye-type strainers will not increase the probability for an accident, increase the consequences of an accident, or create an accident of type not previously analyzed for the following reasons.

1. The new strainers will meet or exceed the ability of the original strainers to function as a system pressure boundary.
2. Seismic qualification of the system has been maintained.
3. The baskets of the new strainers have perforations which provide equivalent protection from foreign matter.
4. Pressure drop across the strainer will be similar; therefore, flow to the condenser section of the Control room Chillers will not be affected.

DCP 92-363

Replace EDG Pressure Gauges on 1-EG-TK-1HA/HB/JA/JB
NAPS - Unit 1

DESCRIPTION

Pressure Indicators 1-EG-PI-614HA/HB and 1-EG-PI-614JA/JB on Emergency Diesel Generator air receivers 1-EG-TK-1HA/HB and 1-EG-TK-1JA/JB had a range of 0 to 600 psig. New gauges were installed having a range of 0 to 300 psig. This narrower range will allow for more accurate readings of the air receiver pressure.

SUMMARY OF SAFETY ANALYSIS (92-SE-MOD-044)

Pressure Indicators 1-EG-PI-614HA/HB and 1-EG-PI-614JA/JB are local indicators on Emergency Diesel Generator air receivers 1-EG-TK-1HA/HB and 1-EG-TK-1JA/JB respectively. The receivers operate at 220 psig and are equipped with relief valves 1-EG-RV-602HA/HB/JA/JB which are set to release at 250 psig. New gauges with a range of 0 to 300 psig were installed. These gauges meet the design requirements of the system. The seismic integrity of the emergency diesel air start system is not impacted by this replacement. Since the function and operation of the emergency diesel air start system was not changed, this modification did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the UFSAR, create a possibility for an accident or malfunction of a different type than previously evaluated, or reduce the margin of safety as defined in the Tech. Specs. Therefore an unreviewed safety question does not exist.

DCP 92-116

IMPROPER INSTALLATION OF RELIEF VALVE
NORTH ANNA UNIT 1

DESCRIPTION

The relief valve on the nitrogen line to the pressurizer relief tank, 1-SI-RV-100, was installed horizontally. Per the manufacturer's (Lonergan) manual, the relief valves must always be installed in the vertical position. The piping was modified so that the relief valve was installed vertically. Most of the discharge piping from the valve was also removed as the discharge line was originally approximately 2 1/2 foot long and was unsupported. The discharge line vents to containment atmosphere and was not required.

SUMMARY OF SAFETY ANALYSIS (92-SE-MOD-044)

This design change did not create an unreviewed safety question as defined by 10CFR50.59.

A. The reorientation of the relief valve did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

The nitrogen to the PRT is used to mitigate overpressurization of the pressurizer by inerting any hydrogen coming out of solution in the PRT. The piping modification insured that the valve operates per design and made it less susceptible to seat leakage. The probability or consequences of an accident was unaffected as the nitrogen flow to the PRT was unaffected.

B. The implementation of this modification did not create a possibility for an accident of a different type than previously evaluated in the Final Safety Analysis Report.

The design function of the valve was not altered. The valve was reoriented to its correct position and no possibility of an accident of a different type was created.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

The relief valve operates in the same manner by lifting when an overpressure condition exists. The margin of safety was not affected.

DCP 92-245

REPLACEMENT OF MAIN STEAM NON-RETURN BYPASS VALVES NORTH ANNA UNIT 1

DESCRIPTION

The main steam non-return bypass valves 1-MS-NRV-103A, 1-MS-NRV-103B and 1-MS-NRV-103C were worn and an exact replacement was not available. The three Rockwell-Edwards 3" figure 3668MT valves were replaced with Edwards 3" figure B36268MLT5 valves. The new valves are shorter in length by 1.3 inches, weigh 44 pounds less, have a flow coefficient (Cv) of 100 verses 90, have a pressure rating of 1690# verses 1500#, have a stem travel of 2.18" verses 1.875" and take ~33 seconds verses ~28 seconds to stroke.

SUMMARY OF SAFETY ANALYSIS (92-SE-MOD-047)

This design change did not create an unreviewed safety question as defined in 10 CRF 50.59. A failure of the pressure boundary was considered and the probability and consequences of this accident will not be increased. All secondary break analysis remain fully bounding.

The small increase in time (~28 to ~33 seconds) it will take to stroke the new valves closed is not significant. These valves are used during start-up to warm the main steam system and to equalize the pressure across the main steam headers non return valves. There are no technical specifications applicable to these NRV bypass valves.

DCP 92-269
Abandonment of Four In-Core Thermocouples
(1-IC-TE-7, 24, 27, and 31)
North Anna Unit 1

Description

Four in-core, core exit, thermocouples were abandoned in place because they are inoperable. The thermocouples will not be repaired or replaced due to ALARA concerns, inaccessibility, and the probability of damaging additional thermocouples in the core during repair efforts. The Technical Specification requirement of two operable thermocouples per quadrant per train was not impacted by this modification. Abandonment consisted of removing the computer points, for the thermocouples, from the ERFCS and ICCM computer software.

Summary of Safety Analysis (92-SE-MOD-048)

This design change in accordance with DCP 92-269 does not create an "unreviewed safety question" as defined in 10 CFR 50.59.

A. The implementation of this modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

   Four in-core, core exit thermocouples were abandoned in place. The remaining thermocouples provide sufficient data for monitoring core exit temperatures. The in-core thermocouples do not provide any control or protection functions. Abandonment does not prevent the Operators from performing necessary measures to mitigate an accident.

B. The implementation of this modification does not create a possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report.

   The in-core thermocouples do not provide any control or protection functions. Abandonment does not prevent the Operators from performing necessary measures to mitigate an accident. No feedback into protective circuitry is possible. This modification removed the associated computer input points from the ICCM and the ERFCS computers. No other computer input is impacted.

Abandonment of Four In-Core Thermocouples
(1-IC-TE-7, 24, 27, and 31)
North Anna Unit 1

Summary of Safety Analysis (continued)

C. The implementation of this modification does not reduce the margin of safety as defined in any Technical Specification.

The minimum required number of operable thermocouples (Section 3.3.3.6) per Technical Specifications is not impacted. At least five operable thermocouples per quadrant per train are available.

DC 92-276

Main Steam Drain Line Modifications
North Anna Unit 2
(#92-SE-MOD-049)

DESCRIPTION

The major issue associated with this Safety Evaluation was the concern for maintaining the short-term interim allowable stress limits for pipe anchor, 1-SHPD-A-126, (i.e. 1.2F). Long-term stress allowable limits were restored on this pipe anchor trunnion by reinforcing the trunnion when the associated line was safely cooled to permit a required integrally welded attachment. The associated line, 3"-SHPD-5-601-Q3, is a main steam drain line header.

This condition was discovered during a pipe stress evaluation for Deviation Report N-92-1687, in which an analysis for a 55 LBS Furmanite clamp (i.e. engineered clamp) was prepared. The Furmanite clamp was to be used to restore the pressure boundary of the pipe in the vicinity of a pipe leak that was originally reported. Subsequent NDE revealed that the pipe wall was thinned out in this area as well.

The installation of the spring hanger was implemented without any Operation's tag-outs. However; the spring hanger hot/cold load settings have been calculated with the 55 LBS Furmanite clamp in place. Upon clamp removal, then a re-distribution of load will occur on the line, requiring the hot/cold load settings for the spring hanger to be re-adjusted. Re-distribution of load will not be great since the total weight of the Furminite clamp is less than 10% of the hot load setting (i.e. 55 LBS vs. 61 LBS), however; 'as-left' settings were set as close to the design setting as could reasonably be achieved. The spring hanger permitted the trunnion to meet short-term interim allowable stress limits (i.e. 1.2F) until the trunnion was reinforced. Despite the fact that short-term allowable stresses were exceeded in the pipe anchor trunnion, no catastrophic failure of the main steam drain line header was expected.

Calculation number CE-0956 has been prepared to justify the addition of the spring hanger on line 3"-SHPD-5-601-Q3. All local details of the spring hanger have been discussed in calculation number DEO-0124. The above mentioned leak in the line was repaired with a Furmanite clamp and the damaged line was replaced via ASME Section XI Repair/Replacement Program. A re-verification of the hot/cold lad settings was performed to ensure that settings are as close to the design setting as can reasonably be achieved.

DC 92-276

Main Steam Drain Line Modifications
North Anna Unit 2

SUMMARY OF SAFETY ANALYSIS

In as much as these proposed modifications were implemented under standing NSS work procedures, without any special Operations tag-outs, restored the acceptable stress levels in all components, and are justified by engineering calculations; the probability of occurrence for the NAPS UFSAR Chapter 15 accidents.

In as much as these proposed modifications were implemented under standing NSS work procedures, without any special Operations tag-outs, restored the acceptable stress levels in all components, and are justified by engineering calculations; the consequences of previously defined NAPS UFSAR Chapter 15 accidents will not be increased.

In as much as these proposed modifications were implemented under standing NSS work procedures, without any special Operations tag-outs, restored the acceptable stress levels in all components, and are justified by engineering calculations; no known potential exists for exists for an accident of a different type than was previously defined in NAPS UFSAR Chapter 15.

DCP-91-007-1
PRESSURIZER PRESSURE TRANSMITTER REPLACEMENT
NORTH ANNA POWER STATION
UNIT 1

DESCRIPTION

In order to reduce the Channel Statistical Allowance associated with harsh environmental conditions, the pressurizer pressure transmitters were replaced by a transmitter with a higher accuracy during harsh environmental conditions. Each Rosemount 1153 Series D transmitter for Loops P-2455, 2456, and 2457 was replaced with a Rosemount 1154 Series H transmitter.

The Channel Statistical Allowance for harsh environmental conditions is reduced by using the Rosemount 1154 Series H transmitter, making it feasible for the pressurizer pressure low SI function to be obtained. Even with the lower Channel Statistical Allowance the setpoint of 1765 psig for pressurizer pressure low SI was raised to 1780 psig (in the conservation direction) for actuation.

SUMMARY OF SAFETY ANALYSIS (92-SE-MOD-050)

An unreviewed safety question relative to this modification does not exist based on the evaluation summarized below.

Safety injection on low-low pressurizer pressure plays a role in large and small LOCAs and steamline breaks, therefore, these accidents were considered in the evaluation. The proposed change affected only accident mitigating systems, therefore it will not increase the frequency of the accidents considered. Since the change ensured the actuation of an accident mitigating function, it will not increase the consequences of the accidents considered. This change enhanced the performance of an existing design, therefore it will not create the possibility for an accident of a different type than previously evaluated.

The modification affects components in safety injection channels, therefore this evaluation considered the impact on spurious operation of the safety injection system at power. Since the circuitry will not be altered and the replacement transmitters have a reliability as good or better than the existing units, the probability of occurrence of a spurious safety injection will not be increased by this change. Should such a malfunction occur, the consequences will be the same as currently described in the UFSAR. The proposed change has provided added assurance that an existing design will perform as intended, therefore no new possibilities for equipment malfunction will be created.
The proposed operational setpoint is more conservative than

previously implemented, therefore safety margins are preserved. Implementation of this more conservative operational setpoint did not require a change to the Technical Specifications. The proposed operational setpoint of 1780 psig meets the existing Technical Specifications which require this setpoint to be $\geq$ 1765 psig.

DCP-91-008-2
PRESSURIZER PRESSURE TRANSMITTER REPLACEMENT
NORTH ANNA POWER STATION
UNIT 2

DESCRIPTION

In order to reduce the Channel Statistical Allowance associated with harsh environmental conditions, the pressurizer pressure transmitters were replaced by a transmitter with a higher accuracy during harsh environmental conditions. Each Rosemount 1153 Series D transmitter for Loops P-2455, 2456, and 2457 was replaced with a Rosemount 1154 Series H transmitter.

The Channel Statistical Allowance for harsh environmental conditions is reduced by using the Rosemount 1154 Series H transmitter, making it feasible for the pressurizer pressure low SI function to be obtained. Even with the lower Channel Statistical Allowance the setpoint of 1765 psig for pressurizer pressure low SI was raised to 1780 psig (in the conservation direction) for actuation.

SUMMARY OF SAFETY ANALYSIS (92-SE-MOD-050)

An unreviewed safety question relative to this modification does not exist based on the evaluation summarized below.

Safety injection on low-low pressurizer pressure plays a role in large and small LOCAs and steamline breaks, therefore, these accidents were considered in the evaluation. The proposed change affected only accident mitigating systems, therefore it will not increase the frequency of the accidents considered. Since the change ensured the actuation of an accident mitigating function, it will not increase the consequences of the accidents considered. This change enhanced the performance of an existing design, therefore it will not create the possibility for an accident of a different type than previously evaluated.

The modification affects components in safety injection channels, therefore this evaluation considered the impact on spurious operation of the safety injection system at power. Since the circuitry will not be altered and the replacement transmitters have a reliability as good or better than the existing units, the probability of occurrence of a spurious safety injection will not be increased by this change. Should such a malfunction occur, the consequences will be the same as currently described in the UFSAR. The proposed change has provided added assurance that an existing design will perform as intended, therefore no new possibilities for equipment malfunction will be created.
The proposed operational setpoint is more conservative than

previously implemented, therefore safety margins are preserved. Implementation of this more conservative operational setpoint did not require a change to the Technical Specifications. The proposed operational setpoint of 1780 psig meets the existing Technical Specifications which require this setpoint to be $\geq$ 1765 psig.

DCP 92-229

INSTALL NIPPLE IN EDG WALL FOR
OIL LINE PENETRATIONS
NORTH ANNA UNIT 1&2

DESCRIPTION

The lube oil equalizer line for the emergency diesels, passes through the diesel doghouse wall connecting the lube oil strainer and filter. Where this line penetrates the wall, it was subjected to vibrations which caused constant wear due to the sharp edges of the wall. A brass nipple was installed through the wall and the oil line was run through it.

SUMMARY OF SAFETY ANALYSIS (92-SE-MOD-051)

This design change did not create an unreviewed safety question as defined by 10CFR50.59.

A. The installation of the nipple in the doghouse wall did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

The nipple is outside of the system boundaries and does not affect the ability of the diesel to operate.

B. The implementation of this modification did not create a possibility for an accident of a different type than previously evaluated in the Final Safety Analysis Report.

The modification did not affect the function or operation of the diesels or their lube oil systems. The EDG operates as designed.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

The emergency diesels still operate as designed. The installation of the nipples in the doghouses did not affect the ability of the emergency diesels to perform their safety function.

DC 90-06-1
Steam Generator Primary Coolant Pipe Whip Restraint Removal
North Anna Unit 1

DESCRIPTION

In support of the Steam Generator (S/G) repair project, the majority of the hot leg pipe whip restraint located within a high radiation area, directly under each S/G, has been permanently removed from containment. The removal of these restraints allowed maximum accessibility to the area while involving the least amount of steel structure removal and personnel exposure. Among the benefits experienced during the S/G replacement were lower man-rem exposure and an increase in general area safety due to reduced congestion underneath the S/Gs.

The entire upper portion of the hot leg pipe whip restraint steel structure and the upper shim section of the crossover leg pipe whip restraint structure has been removed. The lower portion of each restraint assembly (base frame weldment) remains in place. Low personnel exposure time was accomplished by limiting the work activities to cutting at preselected locations having minimum cross-sectional areas.

SUMMARY OF SAFETY EVALUATION - 92-SE-MOD-053

All primary coolant pipe whip restraints are no longer required per the NRC approved Westinghouse leak-before-break (LBB) analysis (WCAPs 11163/11164) and the October 27, 1987 amendment to General Design Criteria 4 (GDC 4). An unreviewed safety question does not exist as a result of this Design Change for the following reason. This Design Change did not adversely affect any accidents evaluated in the Safety Analysis Report (SAR) per the amendment to GDC-4. These bumpers were considered in the subcompartment pressurization analysis. The current NRC philosophy concerning the LBB analysis for the North Anna reactor coolant loops allows the removal of all primary coolant pipe whip restraints without reevaluating subcompartment pressurization analyses. Per the above amendment to GDC-4: "Dynamic effects of pipe rupture covered by this rule are missile generation, pipe whipping, pipe break reaction forces, jet impingement forces, decompression waves within the ruptured pipe and dynamic or nonstatic pressurization in cavities, subcompartments, and compartments. However, cavities, subcompartments and compartments necessary to the containment function are not affected by this modification"... to the GDC-4 ruling. The static subcompartment pressurization effects are insignificant with respect to the existing design basis dynamic pressurization. the whip restraints were not considered in the containment integrity (function) analysis and therefore the containment integrity analysis is unaffected by this modification. The margin of safety of any Technical Specification has not been reduced nor is there a change to a Technical Specification required.

D 90-13
STEAM GENERATOR REPLACEMENT
NORTH ANNA UNIT 1

DESCRIPTION

Due to the degradation of the previous steam generator tubing, the lower steam generator tube bundle assemblies have been replaced at North Anna Power Station Unit 1. The new steam generator lower assemblies were fabricated in accordance with ASME Code Section III, 1986 Edition and have physical, mechanical, and thermal characteristics that are consistent with the original design and safety analysis presented in the Updated Final Safety Analysis Report (UFSAR). The new steam generator lower assemblies are designed and fabricated to be physical duplicates of the original lower assemblies since all major external dimensions and orientation angles for both the original and new components are essentially the same.

Certain design changes and enhancements have been made in the new steam generator lower tube bundle assemblies which address the operating experience of the original steam generators and which enhance the overall reliability and maintainability of the steam generators. These changes and enhancements do not adversely affect the mechanical or thermal-hydraulic performance of the new steam generators.

Specifically, some of these enhancements are the utilization of thermally-treated alloy 690 tubing to reduce the susceptibility of stress and intergranular corrosion experienced by the previous mill-annealed alloy 600 tubing. In addition, the incorporation of an additional row of anti-vibration bars uniformly inserted into the tube bundle provides increased support in the tube bundle region, reducing the susceptibility of the tubes to vibration. The number of tubes has also increased for additional plugging margin.

The steam generator replacement was performed in accordance with the requirements of the ASME Code, Section XI, 1983 Edition and Summer 1983 Addenda. Welding, postweld heat treatment, nondestructive examination, and baseline inservice inspection were performed in accordance with the ASME Code, Section XI, 1983 edition; ASME Code, Section III, 1986 edition; and ANSI B31.7, 1969 edition through 1970 addenda, as applicable.

The steam generator lower assemblies were be removed and replaced through the existing containment equipment hatch. This replacement process is commonly referred to as the two-piece replacement method. The two-piece replacement through the equipment hatch was determined to be the best overall method for North Anna Unit 1 due to limitations on the diameter of equipment that can be moved through the equipment hatch. The containment equipment hatch is large enough to allow passage of the steam generator lower assemblies only, not the steam domes. The previous SG lower assemblies were removed by severing the reactor coolant and all other attached piping at the steam generator nozzles, severing the steam generators within the transition cone, and removing the lower assemblies from the containment. Following removal of the old lower assemblies, the new shop-fabricated steam generator lower assemblies were transported into the containment through the equipment hatch and connected to the original steam domes and reactor coolant piping. Use of the equipment hatch eliminated the need to modify the containment wall concrete or pressure boundary.

Additional enhancements resulting from the replacement of the steam generators are:

- The steam generator blowdown nozzles coupling size has increased to 2-1/2". The previous 2" blowdown piping from these nozzles to the 3" headers has been replaced with 2-1/2" chrome-moly steel pipe for increased erosion/corrosion resistance. Additionally, the previous 1" carbon steel drain line has been replaced with 1" chrome-moly material. These modifications will provide for additional blowdown capability in the future.

- The feedwater piping loop seal at the steam generator has been be replaced with chrome-moly material. This piping has been replaced to alleviate future erosion/corrosion concerns.

- The steam generator and adjacent piping has been covered with new blanket insulation which has stainless steel jacketing. The replacement blanket insulation meets or exceeds the design requirements of the previous insulation.

- The steam generator upper restraints have been replaced with an equivalent restraint. Demolition of the previous upper restraints was required due to difficulties associated with their removal from the steam generators and the effort involved to reuse them.

SUMMARY OF SAFETY EVALUATION - 92-SE-MOD-054

The Safety Evaluation addresses eight major activities associated with steam generator replacement as follows:

I. Steam Generator Vessel Repair
II. Piping Removal and Replacement
III. Instrumentation Removal and Replacement
IV. Blowdown System Modification
V. Insulation Removal and Replacement
VI. Rigging Activities Inside the Containment
VII. Rigging and Transport of Heavy Loads Outside the Containment

VIII. Temporary Services (Impact on permanent facilities only)

I. Steam generator vessel repair constitutes replacement of all three steam generator lower assemblies, including replacement of tubes, tubesheet, lower vessel shell, channelhead, and a portion of the wrapper plate and transition cone. The repair also includes the installation of a flow restrictor in the main steam nozzle and removal of the downcomer flow resistance plates, replacement of the steam generator upper lateral restraints, and new materials for the steam generator lower supports.

II. Piping removal and replacement includes all piping systems attached to the steam generators. These systems include the reactor coolant, main steam, feedwater, chemical feed, wet lay-up, and sample piping which were severed at the steam generator nozzles to allow for the removal of the original steam generator lower assemblies and the installation of the new lower assemblies. The severed piping was reinstalled in essentially the same configuration. Material upgrades from carbon steel to chrome-moly were utilized on the feedwater loop seals for improved erosion/corrosion characteristics. In addition, decontamination of the reactor coolant system piping following the severance cuts is addressed.

III. The steam generator level instrument piping and tubing was severed to allow removal of the original steam generator lower assemblies and installation of the new lower assemblies. The condensate pots and instrument root valves were removed and replaced. The severed piping and tubing as well as the condensate pots were reinstalled to satisfy the original design requirements with material upgrades. In addition, the optical templating bracket installed under DC 92-006-1 was removed.

IV. The previous 1 and 2 inch carbon steel blowdown lines connected to each steam generator lower assembly were replaced with new 1 and 2 1/2 inch chrome-moly lines respectively. The supports associated with the piping were removed and modified/replaced.

V. The original steam generator insulation, of which part was reflective and part was encapsulated fiberglass, was replaced with a blanket-type of insulation that exhibits equivalent thermal properties.

VI. Lifting and handling activities required to support removal and installation of steam generator lower assemblies were evaluated.

VII. Erection, operation, load test, and disassembly of the outside lifting system adjacent to the containment equipment hatch. Establishing and testing the proposed haul route to be used to transport the new steam generators to the equipment hatch, transport the old steam generators to the Old Steam Generator Storage Facility, and transport other heavy loads to and from the containment.

VIII Temporary modifications to support steam generator replacement were required. These temporary modifications included attachment of a flexible duct and volume control damper to the purge system, modification to RCP-1B power supply for temporary steam generator replacement power, modification to security door A-95-1, an auxiliary crane, a jib crane, temporary main steam work platforms, and a reactor cavity cover.

The probability of occurrence for the accidents previously identified have not been increased as discussed below.

I. The probability of accident occurrence associated with the replacement of the SGs has not increased because the design, materials, and code standards for installation are equal to or more conservative than those used in the original licensing basis.

II. All reactor coolant system and secondary side piping and supports have been restored to their original design configuration in accordance with ASME Section XI and ANSI B31.7 code requirements. Replacement materials, including all weld metal utilized, satisfy the original code requirements and meet the existing installation specification. All modified piping systems were subjected to nondestructive examination and hydrostatic testing in accordance with the Section XI, the Special Processes Manual (DC 90-13-1, Reference 6.5) and DC 90-13-1, Appendix 4-21, as applicable. In addition, periodic inspection will continue throughout the remaining life of the plant.

III. The piping, instrument tubing, and condensate pots were reinstalled to satisfy the original design requirements with material upgrades. The piping, instrument tubing, condensate pots, root isolation valves, and vent valves were replaced with a material which meets or exceeds existing material characteristics. Therefore, the probability of an accident has not increased from the original licensing basis.

IV. The modification to the steam generator blowdown system enhances the reliability of the blowdown system with respect to erosion/corrosion concerns. All supports associated with the blowdown system modifications have been reviewed to ensure that the blowdown system meets the original seismic design requirements. The improvements made to the blowdown system meet or exceed the current licensing basis requirements and do not increase the probability of occurrence of an accident.

V. The replacement blanket insulation has been procured and installed to meet or exceed the original design requirements for heat transfer and has been seismically qualified. Failure of the blanket insulation cannot initiate any of the relevant accidents addressed by this design change. Therefore, the issue of probability of accident occurrence is not applicable.

VI. Rigging activities covered by this safety evaluation that may be performed during defueling/refueling were performed in accordance with the existing station heavy loads procedures and in a manner that will not interfere in any way with defueling or refueling operations that may be in progress.

VII. Activities associated with the rigging and transport of heavy loads outside the containment including the steam generator lower assemblies, haul route test load, etc. did not increase the probability of occurrence of an accident.

VIII. The temporary modifications do not have the potential to increase the probability of occurrence of relevant accidents while the unit is defueled.

The modifications addressed in this design change package do not increase the consequences of an accident for the following reasons:

I. As a result of the steam generator vessel repair, Technical Report NE-883, Revision 1 was prepared to consolidate and summarize the safety analyses and evaluations supporting North Anna 1 operation following the steam generator repair. In this report, each UFSAR Chapter 15 accident analysis applicable to North Anna Unit 1 operation with the repaired steam generator has been evaluated and it has been determined that none of the accident consequences were found to be more limiting than those currently documented in the UFSAR.

II. The reactor coolant, main steam, feedwater, wet layup, sample, and chemical feed systems will be restored to their original configuration after SG replacement. As described in approved calculations, all applicable design basis seismic stress and support analyses have been evaluated/performed, as applicable, to verify the capability of the repaired systems to perform their intended functions.

III. The piping, instrument tubing, and condensate pots were reinstalled to satisfy the original design requirements using upgraded material. Therefore, all design basis evaluations of the consequences of accidents for which steam generator level instrumentation is assumed to be operable remain valid.

IV. The function and operation of the steam generator blowdown system did not change. The increase in pipe size does not increase the consequences of an accident since these pipe sizes are bounded by the larger break sizes used in the accident analysis. Therefore, all design basis evaluation of the consequences of accidents involving the blowdown system remain valid.

V. The replacement thermal insulation has been qualified for use within containment and has been procured to meet the post accident environmental conditions within containment. The replacement insulation is seismically installed to ensure the insulation remains attached to the generator in the event of a seismic occurrence. The insulation performs no safety function in the event of a design basis accident. Appropriate evaluations have been performed in accordance with the recommendations of Regulatory Guide 1.82 to ensure that the replacement insulation does not adversely affect emergency core cooling and engineered safeguards systems. Calculations performed in support of the debris analysis (DC 90-13-1, Appendix 4-28, Steam Generator Insulation Debris Analysis Letter), analyze the LOCA with new input resulting from the insulation replacement. The calculations determine that the inside and outside recirculation spray and low head safety injection pumps will perform their safety related functions. This debris analysis has also determined that the insulation debris fragments that pass through both the sump screens and pump suction screens would not affect the operation of the inside and outside recirculation spray and low head safety injection pumps. In addition, the small fragments that pass through the sump and pump suction screens will not cause blockage of the header spray nozzles. Therefore, all design basis evaluations of the consequences of accidents are unaffected by the replacement insulation.

VI. All movements of heavy loads within containment while fuel remain within the reactor containment were conducted in accordance with the existing station heavy loads procedures to ensure that the loads remain within the established safe load paths and to ensure that, in the inadvertent event of a load drop, the consequences remain within the established acceptance criteria. No safety-related equipment would be adversely impacted by a drop outside the containment. However To assess the radiological consequences associated with this drop, an analysis was performed. The acceptability of the offsite dose consequences associated with a postulated drop have been evaluated and compared to the consequences of other events in the same class of postulated accidents for waste gas or waste liquid releases. The evaluated consequences of a steam generator lower assembly drop are within the applicable regulatory guidelines and are less than the limiting, and more permanent, licensing basis accidents currently evaluated in the UFSAR. Thus, the consequences associated with this class of accidents will not be increased.

VII. Activities associated with the rigging and transport of heavy loads outside the containment including the steam generator lower assemblies, haul route test load, etc. will not increase the consequences of any accidents.

VIII. In the defueled condition, all UFSAR accidents associated with reactor operation, reactor criticality and reactor decay heat removal are not credible occurrences. The only accident which required consideration regarding temporary modifications was the fuel handling accident inside containment. During all fuel handling, containment integrity was maintained. Modification to the containment purge system was not made until the vessel was defueled. The temporary main steam work platforms were installed in accordance with the Station Heavy Loads Procedure (0-MCM-1303-01).

The possibility for an accident of a different type than was previously evaluated in the Safety Analysis Report has not been increased as justified below:

I. The possibility of an accident that is different from that already evaluated in the UFSAR is not created because, as evaluated in Westinghouse Safety Evaluation SECL-90-113, the replacement steam generators have been designed and fabricated to criteria that are equivalent to or better than the existing steam generators. The replacement steam generators have also been determined to have no adverse impact on the function or performance of connected systems, components and structures. The upper restraints were not removed prior to fuel offload and were reinstalled prior to refueling.

II. All reactor coolant system and secondary side piping and supports were restored to their original configuration in accordance with the original code requirements using materials which meet or exceed the original design requirements. Various configurations of primary and secondary pipe cuts with fuel in the pool were evaluated and seismic supports specified where required to maintain a seismicly acceptable system.

III. The piping, instrument tubing, were reinstalled to satisfy the original design requirements using upgraded material. The function and operation of the system following the modification did not change. Therefore, the possibility of an accident of a different type from that evaluated previously would not be created.

IV. The blowdown piping changes implemented by this modification did not change the function and operation of the steam generator blowdown system. Therefore, the possibility of an accident of a different type from that evaluated previously would not be created.

V. The replacement insulation performs the same function as the original insulation and is seismically installed. Failure of the SG insulation does not, in itself, initiate any existing type of accident. The change from encapsulated insulation to blanket insulation does not alter this conclusion. Therefore, no new accident is possible as a result of the replacement insulation.

VI. Rigging activities performed during defueling operations were conducted in accordance with the existing heavy load handling procedures to ensure that load handling occurs only in currently analyzed and approved safe load paths. The radiological consequences of a postulated drop of an old steam generator lower assembly inside the containment or within the protected area have been evaluated and determined to be within applicable regulatory limits and less than the limiting case events within the same classification of accidents currently evaluated in the UFSAR. Thus, no new accidents are created as a result of the rigging activities.

VII. Activities associated with the rigging and transport of heavy loads outside the containment including the steam generator lower assemblies, haul route test load, etc. did not create the possibility of an accident of a different type than previously evaluated.

VIII. Following the completion of the SGR and prior to a return to power, all temporary modifications were removed. Thus, this activity did not create the possibility of an accident of a different type as the operating performance of the plant following SGR is identical to the operating performance before SGR.

The margin of safety as defined in the basis for any Technical Specification is not reduced by this design change. The replacement steam generators have been demonstrated to insignificantly affect the transient system response during postulated UFSAR Chapter 15 accidents. Accident analyses for all UFSAR Chapter 15 transients have been performed which bound allowable operation in accordance with the North Anna 1 Technical Specifications that will be applicable following steam generator replacement. All accident analyses meet their respective acceptance criteria. It may, therefore, be concluded that steam generator replacement does not decrease the margin of safety as defined in the basis for any Technical Specification.

No revision to the Technical Specification is needed as a result of this design change. Technical Specification Amendments 153 and 154 were issued to allow operation at reduced power levels until completion of the steam generator lower assembly replacement. These technical specification changes have reverted to their pre-reduced power level values following completion of the steam generator replacement.

DESIGN CHANGE PACKAGE 91-193
ADDITION OF P-4 TURBINE TRIP TEST POINTS
NORTH ANNA UNIT 1

DESCRIPTION

Two new terminal blocks, equipped with banana jack adapters, were installed in the reactor trip switchgear cabinet 1-EI-CB-46A. These terminal blocks were wired to points located in the back of the cabinet, which were used by station electricians to determine breaker position during the monthly PT on the SSPS. The new blocks which are located at the front of the cabinet allow easier access to the test points.

SUMMARY OF SAFETY ANALYSIS (92-SE-MOD-055)

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

A. The implementation of this modification did not increase the probability of occurrence or consequences of an accident or malfunctions of equipment important to safety as previously evaluated in the UFSAR.

   Accident probability has not been increased because this design change conformed to standards and admins. The terminal block installation is an electrical extension of an existing circuit. The installation will not impair or degrade any reactor trip switchgear. This DCP decreases the likelihood of causing an inadvertent unit trip, and does not impair any safety system.

B. The implementation of this modification did not create a possibility for an accident or a malfunction of a different type than any previously evaluated in the UFSAR because the design change is minor and will not affect the operation of the Reactor Trip System.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification because neither of the logic trains will be impaired by this modification.

DCP 92-241

VARIOUS VALVE REPLACEMENTS DURING SGR PROJECT
NORTH ANNA UNIT 1

DESCRIPTION

This DCP controlled the replacement of three 3" Reactor Coolant System drain valves (1-RC-29, 1-RC-68, 1-RC-100) and three 2" Reactor Coolant system flow bypass isolation valves (1-RC-6, 1-RC-45, 1-RC-77) which were stuck in the open position. The valves were replaced on a one for one basis with Conval globe valves suitable for use in the Reactor Coolant System. The valves were replaced during the Unit 1 Steam Generator Replacement Outage.

SUMMARY OF SAFETY ANALYSIS (92-SE-MOD-057)

The replacement of these six Reactor Coolant System valves on a one for one basis does not constitute an unreviewed safety question or require a modification to the Technical Specifications. The function and operation of the valves and the Reactor Coolant System remain the same. The replacement does not increase the probability, consequence or possibility of an accident. The Margin of Safety as set forth in the Tech Specs. is not affected. Accidents of a different type than previously analyzed are not possible.

DCP 92-015

IPE INTERNAL FLOODING CIVIL MODIFICATIONS
(SE #92-SE-MOD-058)

DESCRIPTIONS

Generic Letter 88-20 "Individual Plant Examination (IPE) for Severe Vulnerabilities - 10 CFR50.54(f) required that utilities perform a systematic study to identify vulnerabilities to various severe accidents. The IPE conducted for North Anna recommended that several modifications be made to reduce the potential for core damage due to internal flooding. The modifications performed by this DCP are as follows.

1. Reinforced the firestops between the Quench Spray Pumphouse and the Auxiliary Building so that a hydrostatic head of 17.66 feet could be resisted.
2. Installed flood barrier between the Chiller Room and the Emergency Switchgear Room. The barrier consists of walls 3.25 feet high.
3. Modified doors between Turbine Building and Chiller Room so that flood waters would flow from the Chiller Room to the Turbine Building.

SUMMARY OF SAFETY ANALYSIS

The safety evaluation determined the main concern was to reduce the plant's vulnerability to flood induced events. The modifications decreased the probability of core damage from flood related events. The safety evaluation concluded that an unreviewed safety question does not exist.

DCP 92-299

REMOVAL OF UNUSED LOOP STOP VALVE DISC
PRESSURIZATION LINE VALVES DURING SGR PROJECT
NORTH ANNA UNIT 1

DESCRIPTION

Five existing Reactor Coolant System loop stop valve disc pressurization line valves, 1-RC-184, 1-RC-188, 1-RC-191, 1-RC-193, and 1-RC-196 were no longer being used. A similar valve had leaked in the past. To avoid a potential leakage problem on the remaining five valves, the valves were removed by this DCP and the 3/4" lines were capped to maintain the system pressure boundary. The valves were replaced during the Unit 1 Steam Generator Replacement Outage.

SUMMARY OF SAFETY ANALYSIS (92-SE-MOD-060)

The removal of these five Reactor Coolant System loop stop valve disc pressurization line valves does not constitute an unreviewed safety question or require a modification to the Technical Specifications. The valves were no longer used and represented a potential leakage problem. The function and operation of the loop stop valves and the Reactor Coolant System remain the same. The replacement does not increase the probability, consequence or possibility of an accident. The Margin of Safety as set forth in the Tech Specs. is not affected. Accidents of a different type than previously analyzed are not possible.

DCP 92-324

REPLACEMENT OF 1-MS-80 DURING SGR PROJECT
NORTH ANNA UNIT 1

DESCRIPTION

1-MS-80 is a 1½" Main Steam line vent valve. The original valve was leaking through and required replacement. Therefore, the valve was replaced with a Conval model 1.5-11G2C-1056H globe valve during the Steam Generator Replacement Outage.

SUMMARY OF SAFETY ANALYSIS (92-SE-MOD-064)

The replacement of the 1½" vent valve, 1-MS-80, with a Conval model 1.5-11G2C-1056H globe valve does not constitute an unreviewed safety question or require a modification to the Technical Specifications. The function and operation of the valve and the Main Steam System remain the same. The replacement does not increase the probability, consequence or possibility of an accident. The Margin of Safety as set forth in the Tech Specs. is not affected. Accidents of a different type than previously analyzed are not possible.

DCP 92-277

RE-ROUTING OF VENT LINE - 1/2" SI-691-ICN8-Q2
NORTH ANNA, UNIT 2
(92-SE-MOD-065)

DESCRIPTION

Vent line 1/2"-SI-691-ICN8-Q2 downstream of isolation vent valve 2-SI-377 is re-configured and its support system modified to provide more flexibility so as to accommodate for the sudden transient movement of the 8"-SI-449-153A-Q2 line resulting from the LHSI pump start. The modification involves providing an expansion loop on the horizontal of the vent tubing as well as alter/remove two restraint mode points on the vent tubing span.

SUMMARY OF SAFETY ANALYSIS

This modification did not create an "unreviewed safety question" as defined in 10CFR 50.59.

The implementation of this modification did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment to safety as previously evaluated in the Final Safety Analysis Report.

Vent line 1/2"-SI-691-ICN8-Q2 downstream of isolation vent valve 2-SI-377 as well as its supports were analyzed for the sudden transient movement of 1/2" (in East direction) of the 8" safety injection line as well as dead weight, thermal & seismic loadings. Thus, this modification has not only accounted for design basis seismic loads but also taken into account transient loads occurring as a result of the sudden movement of the 8"-SI line.

The implementation of this modification does not create a possibility for an accident or malfunction of a different type than any previously evaluated on the Final Safety Analysis Report.

As previously stated, this modification has not only accounted for the seismic loading but has also evaluated for the sudden transient movement/loads of the 8" safety injection line upon LHSI pump start. Possibility of a different type of accident is not created as a result of this modification.

The implementation of this modification does not reduce the margin of safety as defined in any Technical Specification.

Reconfiguration of the vent line down stream of Isolation Vent Valve 2-SI-277 will still keep the boration flow paths on the 8"-SI-449-153A-Q2 line operational during and after the modification, thus the Technical Specifications are not affected.

DCP 91-009

REPAIR/REPLACEMENT OF 24" SERVICE WATER HEADER TO/FROM UNIT 1 RSHX'S

DESCRIPTION

The Service Water System at North Anna was built using mostly carbon steel components. Corrosion of the carbon steel has been a problem. Several Design Changes have been implemented to repair and/or replace portions of this system.

This DCP repaired and partially replaced the deteriorated 24" diameter Service Water pipes to the Unit 1 Quench Spray Building which serve the Unit 1 RSHXs. The repair/replacement was implemented during the Unit 1 steam generator replacement outage. While the repair work was in progress, the 24" headers to the Unit 1 RSHXs were temporarily blanked off from the rest of the system. This allowed the Unit 2 Service Water system and those portions of the system that are common to both units to remain in service.

The 24" headers below the Service Building were repaired. Repair consisted of cleaning the inside of the piping, weld repair where required, and application of a coating system.

The buried pipe between the Service Building and Quench Spray Building was replaced. The horizontal portion of the pipe below the nonexcavated area was replaced with 22" diameter pipe utilizing the existing pipe as a sleeve. The replacement piping was internally coated in the same manner as the piping that was repaired. Buried piping that was replaced was protected from external corrosion by tape wrapping.

To provide cooling water to the Unit 1 Control Room Chiller while the repair/replacement of the 24" headers was in progress, a temporary supply from the Bearing Cooling Water System was provided.

SUMMARY OF SAFETY ANALYSIS (92-SE-OT-066)

Implementation of this DCP did not involve an unreviewed safety question:

The excavation of the buried service water lines exposed the lines and some electrical conduit ducts to possible hazards which may interrupt power to critical equipment. The normal design basis protection against natural phenomena afforded by the earth and concrete was temporarily removed during the excavation. The NRC granted a temporary exception to 10CFR Part 50, Appendix A, "General Design Criteria (GDC) for Nuclear Power Plants." Specifically, Criteria 2 (GDC-2) requires design protection against the effects of natural phenomena (e.g., earthquakes, tornadoes) for components, systems and structures important to safety.

During the restoration work, each Service Water header was isolated and partially drained to install and remove code piping plugs on the Unit 1 Service Water lines. Unit operation at power with a single train of service water is allowed for periods of up to seven days by plant technical specifications. However, this isolation resulted in a loss of redundancy in the Service Water system for six periods of up to seven days. A PRA of the project concluded that performing the excavation and backfill activities while Unit 1 is operating resulted in a negligible (<1E-6) contribution of the CDF and the probability of a Core Damage (CD) event occurring during these periods. This conclusion was based on the assumption that the Service Water pipes and cable ducts were temporarily supported as required to maintain their seismic qualification and appropriate measures were in place to prevent construction mishaps. The PRA results also indicate that the repeated isolation of service water headers had a small effect (5.1 E-6) on the probability of a CD event for Unit 2. Planning recovery measures to restore the Service Water header further reduced the probability of a CD event during periods of operation with one Service Water header isolated to <1E-6.

DCP 92-328

INSTALLATION OF PERSONNEL AIRLOCK HINGE/DOOR RESTRAINING DEVICE NORTH ANNA UNIT 1

DESCRIPTION

Restraining brackets and shims were installed on the hinges of the unit 1 containment personnel airlock. The hinges are designed to be adjustable to provide a means of aligning the door to obtain a proper seal. The extreme weight of the door caused the hinges to slip, and alignment was lost. These restraining brackets and shims, provided by Holtec International, secure alignment once the door has been adjusted.

SUMMARY OF SAFETY ANALYSIS - (92-SE-MOD-069)

This design change in accordance with DCP 92-328 does not create an "unreviewed safety question" as defined in 10 CFR 50.59.

A. The implementation of this modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

   During containment pressurization, the airlock doors are secured closed by the locking ring, not the restraining brackets. The brackets and shims only maintain the alignment of the door to improve conditions for a proper seal, and actually decrease the probability of leakage.

   The brackets and shims cannot increase the consequences of an accident because they are exterior to the pressure boundary, and do not increase the amount of adjustment to align the door. In other words, the door cannot fall further out of the alignment as a result of this modification.

B. The implementation of this modification does not create a possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report.

   No other equipment was involved in this modification. The modification was very simple in that a shim was placed inside of the hinge bearing, and a bracket was attached to the hinge pillow block and the door mounting block.

C. The implementation of this modification does not reduce the margin of safety as defined in any Technical Specification.

The installation of the split shims and restraining brackets has improved the seal, and will prevent loss of alignment of the door over long periods of time. This modification has actually increased the margin of safety. Periodic Test 1/2-PT-62.1 has been performed to verify that the airlocks meet the Technical Specification leakage requirements.

DCP 92-329

INSTALLATION OF PERSONNEL AIRLOCK
HINGE/DOOR RESTRAINING DEVICE, UNIT 2
(SE #92-SE-MOD-069)

DESCRIPTION

The existing Containment airlock doors had the potential to drop down when the door was opened. This may have caused the door not to line up and seal properly. A restraining device was designed by the airlock vendor which was installed on the inner and outer doors. The retraining device aids in proper alignment of the airlock doors.

SUMMARY OF SAFETY ANALYSIS

The safety evaluation concluded that an unreviewed safety question does not exist.

DC 92-280

Provide GL 89-10 Wiring Modifications
North Anna / Unit 1

Description

The 1-CC-MOV-100A and B motor operators were modified by relocating the torque switch bypasses from rotors 1 and 2 to rotors 3 and 4 along with setting up the bypass rotors for bypass of 20-25% in their respective directions. This was accomplished with a relatively minor wiring modification supplemented with testing and setup.

Summary Of Safety Analysis (92-SE-MOD-070)

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because the performance characteristics of the MOVs (i.e. torque, stroke time, pressure ratings etc.) will not be altered. The torque switch bypass will improve motor operator reliability by reducing the chances of inadvertent trip during valve stroke.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because the application of torque switch bypass using the third and fourth rotors is a common component application for this station and the industry. It does not place any unusual performance demands on the component or systems involved.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because the MOV operation is not altered and its contribution to the safety of the plant is not degraded. Thus the Tech Spec bases are not altered.

DC 92-281

Provide GL 89-10 Wiring Modifications
North Anna / Unit 2

Description

The 2-CC-MOV-200A and B motor operators were modified by relocating the torque switch bypasses from rotors 1 and 2 to rotors 3 and 4 along with setting up the bypass rotors for bypass of 20-25% in their respective directions. This was accomplished with a relatively minor wiring modification supplemented with testing and setup.

Summary Of Safety Analysis(92-SE-MOD-070)

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because the performance characteristics of the MOVs (i.e. torque, stroke time, pressure ratings etc.) will not be altered. The torque switch bypass will improve motor operator reliability by reducing the chances of inadvertent trip during valve stroke.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because the application of torque switch bypass using the third and fourth rotors is a common component application for this station and the industry. It does not place any unusual performance demands on the component or systems involved.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because the MOV operation is not altered and its contribution to the safety of the plant is not degraded. Thus the Tech Spec bases are not altered.

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DC 92-282

Provide GL 89-10 Wiring Modifications
North Anna / Unit 1

Description

The 1-SW-MOV-101A,B,C,D, 1-SW-MOV-105A,B,C and D motor operators were modified by relocating the torque switch bypasses from rotors 1 and 2 to rotors 3 and 4 along with setting up the bypass rotors for bypass of 80-85% in the open direction and 20-25% bypass in the close direction. This was accomplished with a relatively minor wiring modification supplemented with testing and setup.

Summary Of Safety Analysis (92-SE-MOD-070)

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because the performance characteristics of the MOVs (i.e. torque, stroke time, pressure ratings etc.) will not be altered. The torque switch bypass will improve motor operator reliability by reducing the chances of inadvertent trip during valve stroke.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because the application of torque switch bypass using the third and fourth rotors is a common component application for this station and the industry. It does not place any unusual performance demands on the component or systems involved.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because the MOV operation is not altered and its contribution to the safety of the plant is not degraded. Thus the Tech Spec bases are not altered.

DC 92-283

Provide GL 89-10 Wiring Modifications
North Anna / Unit 2

Description

The 2-SW-MOV-201A,B,C,D, 2-SW-MOV-205A,B,C and D motor operators were modified by relocating the torque switch bypasses from rotors 1 and 2 to rotors 3 and 4 along with setting up the bypass rotors for bypass of 80-85% in the open direction and 20-25% bypass in the close direction. This was accomplished with a relatively minor wiring modification supplemented with testing and setup.

Summary Of Safety Analysis(92-SE-MOD-070)

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because the performance characteristics of the MOVs (i.e. torque, stroke time, pressure ratings etc.) will not be altered. The torque switch bypass will improve motor operator reliability by reducing the chances of inadvertent trip during valve stroke.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because the application of torque switch bypass using the third and fourth rotors is a common component application for this station and the industry. It does not place any unusual performance demands on the component or systems involved.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because the MOV operation is not altered and its contribution to the safety of the plant is not degraded. Thus the Tech Spec bases are not altered.

DC 92-284

Provide GL 89-10 Wiring Modifications
North Anna / Unit 1

Description

The 1-SW-MOV-102A,B and 1-SW-MOV-106A,B motor operators were modified by relocating the torque switch bypasses from rotors 1 and 2 to rotors 3 and 4 along with setting up the bypass rotors for bypass of 80-85% in the open direction and 20-25% bypass in the close direction. This was accomplished with a relatively minor wiring modification supplemented with testing and setup.

Summary Of Safety Analysis(92-SE-MOD-070)

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because the performance characteristics of the MOVs (i.e. torque, stroke time, pressure ratings etc.) will not be altered. The torque switch bypass will improve motor operator reliability by reducing the chances of inadvertent trip during valve stroke.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because the application of torque switch bypass using the third and fourth rotors is a common component application for this station and the industry. It does not place any unusual performance demands on the component or systems involved.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because the MOV operation is not altered and its contribution to the safety of the plant is not degraded. Thus the Tech Spec bases are not altered.

DC 92-285

Provide GL 89-10 Wiring Modifications
North Anna / Unit 2

Description

The 2-SW-MOV-202A,B and 2-SW-MOV-206A,B motor operators were modified by relocating the torque switch bypasses from rotors 1 and 2 to rotors 3 and 4 along with setting up the bypass rotors for bypass of 80-85% in the open direction and 20-25% bypass in the close direction. This was accomplished with a relatively minor wiring modification supplemented with testing and setup.

Summary Of Safety Analysis(92-SE-MOD-070)

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because the performance characteristics of the MOVs (i.e. torque, stroke time, pressure ratings etc.) will not be altered. The torque switch bypass will improve motor operator reliability by reducing the chances of inadvertent trip during valve stroke.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because the application of torque switch bypass using the third and fourth rotors is a common component application for this station and the industry. It does not place any unusual performance demands on the component or systems involved.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because the MOV operation is not altered and its contribution to the safety of the plant is not degraded. Thus the Tech Spec bases are not altered.

DC 92-286

Provide GL 89-10 Wiring Modifications
North Anna / Unit 1

Description

The 1-SW-MOV-103A,B,C,D, 1-SW-MOV-104A,B,C and D motor operators were modified by relocating the torque switch bypasses from rotors 1 and 2 to rotors 3 and 4 along with setting up the bypass rotors for bypass of 80-85% in their respective directions. This was accomplished with a relatively minor wiring modification supplemented with testing and setup.

Summary Of Safety Analysis (92-SE-MOD-070)

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because the performance characteristics of the MOVs (i.e. torque, stroke time, pressure ratings etc.) will not be altered. The torque switch bypass will improve motor operator reliability by reducing the chances of inadvertent trip during valve stroke.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because the application of torque switch bypass using the third and fourth rotors is a common component application for this station and the industry. It does not place any unusual performance demands on the component or systems involved.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because the MOV operation is not altered and its contribution to the safety of the plant is not degraded. Thus the Tech Spec bases are not altered.

DC 92-287

Provide GL 89-10 Wiring Modifications
North Anna / Unit 2

Description

The 2-SW-MOV-203A,B,C,D, 2-SW-MOV-204A,B,C and D motor operators were modified by relocating the torque switch bypasses from rotors 1 and 2 to rotors 3 and 4 along with setting up the bypass rotors for bypass of 80-85% in their respective directions. This was accomplished with a relatively minor wiring modification supplemented with testing and setup.

Summary Of Safety Analysis(92-SE-MOD-070)

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because the performance characteristics of the MOVs (i.e. torque, stroke time, pressure ratings etc.) will not be altered. The torque switch bypass will improve motor operator reliability by reducing the chances of inadvertent trip during valve stroke.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because the application of torque switch bypass using the third and fourth rotors is a common component application for this station and the industry. It does not place any unusual performance demands on the component or systems involved.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because the MOV operation is not altered and its contribution to the safety of the plant is not degraded. Thus the Tech Spec bases are not altered.

DC 92-288

Provide GL 89-10 Wiring Modifications
North Anna / Unit 1

Description

The 1-SW-MOV-108A and B motor operators were previously modified (EWR 86-552)by relocating the torque switch bypasses from rotors 1 and 2 to rotors 3 and 4 along with setting up the bypass rotors for bypass of 20-25% in their respective directions. In order to be consistent with existing design standards the bypass in the close direction were increased to 80-85%

Summary Of Safety Analysis (92-SE-MOD-070)

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because the performance characteristics of the MOVs (i.e. torque, stroke time, pressure ratings etc.) will not be altered. The torque switch bypass will improve motor operator reliability by reducing the chances of inadvertent trip during valve stroke.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because the application of torque switch bypass using the third and fourth rotors is a common component application for this station and the industry. It does not place any unusual performance demands on the component or systems involved.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because the MOV operation is not altered and its contribution to the safety of the plant is not degraded. Thus the Tech Spec bases are not altered.

DC 92-289

Provide GL 89-10 Wiring Modifications
North Anna / Unit 2

Description

The 2-SW-MOV-208A and B motor operators were previously modified (EWR 86-552)by relocating the torque switch bypasses from rotors 1 and 2 to rotors 3 and 4 along with setting up the bypass rotors for bypass of 20-25% in their respective directions. In order to be consistent with existing design standards the bypass in the close direction were increased to 80-85%

Summary Of Safety Analysis(92-SE-MOD-070)

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because the performance chara teristics of the MOVs (i.e. torque, stroke time, pressure ratings etc.) will not be altered. The torque switch bypass will improve motor operator reliability by reducing the chances of inadvertent trip during valve stroke.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because the application of torque switch bypass using the third and fourth rotors is a common component application for this station and the industry. It does not place any unusual performance demands on the component or systems involved.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because the MOV operation is not altered and its contribution to the safety of the plant is not degraded. Thus the Tech Spec bases are not altered.

DC 92-291

Provide GL 89-10 Wiring Modifications
North Anna / Unit 2

Description

The 2-SW-MOV-210A,B, 2-SW-MOV-213A,B, 2-SW-MOV-214A,B, 2-SW-MOV-217, and 2-SW-MOV-219 motor operators were modified by relocating the torque switch bypasses from rotors 1 and 2 to rotors 3 and 4 along with setting up the bypass rotors for bypass of 80-85% in the open direction and 20-25% bypass in the close direction. This was accomplished with a relatively minor wiring modification supplemented with testing and setup.

Summary Of Safety Analysis(92-SE-MOD-070)

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because the performance characteristics of the MOVs (i.e. torque, stroke time, pressure ratings etc.) will not be altered. The torque switch bypass will improve motor operator reliability by reducing the chances of inadvertent trip during valve stroke.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because the application of torque switch bypass using the third and fourth rotors is a common component application for this station and the industry. It does not place any unusual performance demands on the component or systems involved.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because the MOV operation is not altered and its contribution to the safety of the plant is not degraded. Thus the Tech Spec bases are not altered.

DCP 92-254

ADD TEST SWITCHES
UNDERVOLTAGE RELAYS
NORTH ANNA UNIT 1

DESCRIPTION

On March 6, 1992 an evaluation of surveillance requirements determined that the Unit 2 Reactor Coolant Pump bus monthly undervoltage and underfrequency channel functional tests were not performed monthly in accordance with Technical Specification Table 4.3-1 Items 16 and 17. Although it was possible to perform the monthly functional verification on the undervoltage relays required by Tech Specs, the original physical arrangement did not facilitate any adjustments that may have been required as a result of that testing. This design change involved adding a Westinghouse, Type FT-1, test switch to the RCP Train A and B undervoltage relay circuits for the A, B and C 4KV Buses located in cabinets 1-EP-CB-28E/F, (circuits 1NNSA05, 6 & 7; 1NNSB05, 6 & 7 and 1NNSC05, 6 & 7). This involved installing six (6) switches; three switches were installed in each cabinet. They were mounted on the inner door by cutting three holes as indicated on the drawings. The switches were installed such that the General Electric NGV undervoltage relays are isolated from their respective circuits by the switches. That is, the voltage inputs and the contacts used are removed from the circuits when the switches are opened. These switches are closed during Unit operation, and have covers which must be removed in order to open the switches. The covers are normally sealed by the Control Operations Technicians after functional testing has been performed. This functional testing verifies that the switches have been closed, and that the circuits are functional. Westinghouse "Flexitest" Type FT-1 test switches facilitate the testing of these circuits and minimizes the consequences of human error. The specific test points are provided without being in proximity to other internal wiring and there is no need to use tools or alligator clips. The switch also provides electrical separation from the remainder of the circuit being tested.

SUMMARY OF SAFETY ANALYSIS (92-SE-MOD-071)

This design change did not create an unreviewed safety question as defined in 10CFR50:59.

A. The implementation of this modification did not increase the probability of occurrence of the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Updated Final Safety Analysis Report.

The modification added test switches to the undervoltage circuits which are closed during normal unit operation, and they do not change the functional operation of the undervoltage circuits. This modification improves the safety of handling these circuits both during Unit operation and refuelling outages. These circuits perform their Safety Related functions as they did before the modification.

B. The implementation of this modification did not create the possibility for an accident or malfunction of a different type than evaluated previously in the UFSAR.

The addition of these passive test switches did not change the operation or response of the undervoltage circuits. They operate to perform their functions as they did before the modification, and since they are passive devices they do not introduce any new accident or malfunction that is not already bounded by single failure or common mode analysis.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any technical specification.

The control, function and operating conditions of the undervoltage circuits did not change with respect to their formerly designed safety related functions by the addition of this minor control circuit improvement. The modification in no way affected the availability of the systems for their safe shutdown functions.

DCP 92-306

EDG LOAD SEQUENCING TIMER REPLACEMENT

DESCRIPTION

Certain Agastat 2400/7000 Series timers used in the Unit 1 Emergency Bus Load Sequencing circuits required replacement. These timers were exhibiting setpoint drift which results from the timer's poor repeat accuracy (+/- 10% of setpoint for times greater than or equal to 200 seconds and +/- 5% for times less than 200 seconds). Several Deviation Reports and Licensing Event Reports were written to document the failures. Eight timers selected for replacement, identified in Appendix 4-3, were chosen based on the number of failures and the priority of the timer. The Agastat timers were replaced with Allen-Bradley Type RTC having repeat accuracy of approximately +/- 1% which is well within Technical Specification requirements.

SUMMARY OF SAFETY ANALYSIS: Safety Evaluation # 92-SE-MOD-75

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

A. The implementation of this modification did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Final Safety Analysis Report.

The new timers are performing the same function as the existing equipment. All design parameters have been met or exceeded by the new timers. No modifications to any of the existing input or resultant logic for the timer circuits were altered by this modification.

B. The implementation of this modification did not create the possibility for an accident or malfunction of a different type than evaluated previously in the Final Safety Analysis Report.

Failure of these timers was bounded by single failure criteria. Additionally, manual operator actions to start the associated equipment was not impacted by this modification. All accidents where a loss of off-site power was postulated or an actuation of the ESF functions were considered in this review.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

Replacing the existing timers with Allen-Bradley timers ensured that the requirements for emergency bus load sequencing has been maintained. The new timers perform the same functions as the original timers.

DCP 92-352-1
REDESIGN OF MAKEUP TO CONTROL ROOM
CHILLED WATER SYSTEM

DESCRIPTION

Makeup water for the control Room Chilled Water system was controlled by a pressure control valve, 1-HV-PCV-1303. The pressure control valve had been fouled by foreign matter which would prevent the valve from closing. When this happened the chilled water system became overfilled. When the pressure control valve was not functioning, the operator would have to add makeup water by use of the bypass valve. This was difficult, because the bypass valve was approximately 14′ above the floor.

The pressure control valve and its bypass have been eliminated. The makeup water piping has been rerouted, and a manual valve is used to control makeup to the chilled water system. The manual valve has been located where the operator can reach it from the floor while watching the level glasses in the head tanks. The PVC that was eliminated was designated safety related, but only for its function as a system pressure boundary. The piping, fittings and valve that were use for this modification meet all design requirements of the original installation.

SUMMARY OF SAFETY ANALYSIS (1-SE-MOD-001)

This design change does not create an unreviewed safety question because the ability of the Control Room Air Conditioning Chilled Water System to maintain Control Room Habitability has not been reduced. The chilled water system is a closed system and makeup is seldom required. When makeup is required, the modified piping arrangement will allow the operator to manually add it in a quick and controlled manner. The components that will be used for this modification will meet the design requirements of the original installation. Seismic qualification of the system will be maintained.

DCP 92-327

REPLACEMENT OF RICHMOND INSERTS
WITH MAXI COUPLING BOLTS
NORTH ANNA UNIT 1
(#93-SE-MOD-002)

DESCRIPTION

Some steel Richmond inserts have corroded to a point where they may not be safe for lifting the removable concrete slabs in which they are embedded. A case in point is the removable concrete slab that was dropped in Surry Nuclear Power Stations containment in 1992. The major issue then was the structural integrity of such inserts and the degraded condition that some may be in due mainly to corrosion. A maxi coupling bolt was used to replace the existing Richmond Inserts and features a stainless steel coupling nut flush with the concrete surface.

The replacement was allowed to enhance the structural integrity of the lifting inserts and consequently to preclude dropping of the slabs.

SUMMARY OF SAFETY ANALYSIS

An unreviewed safety question does not exist because the new maxi coupling bolt will adequately carry slab loads. The allowable loads are documented in an addendum to Calculation SEO-1329. There are no changes to the design basis. Overall integrity of the plant is maintained, margins of safety are not decreased, there are no changes to Tech Spec if UFSAR, and consequences and probabilities of accidents are not increased.

SAFETY EVALUATION NUMBER 93-SE-MOD-003

DESCRIPTION

Tubing from Chilled Water flow element 2-HV-FE-2202 will be rerouted between flow element taps and flow indicator 2-HV-FI-2202. This change will increase clearance above a personnel ladder over a flood wall.

SAFETY EVALUATION SUMMARY

The change does not alter any system description or operation. The flow element will still be operational while the tubing modification is implemented. Only local monitoring will be briefly interrupted.

DCP 92-211

REPLACE THE FUEL POOL RULER
(SE #93-SE-MOD-004)

DESCRIPTION

The existing Fuel Pool Ruler was difficult to read therefore Operations requested that new level rulers be installed to facilitate daily Operator inspections. Three rulers were installed at different locations as requested by Operations. The rulers were designed so that they could be easily read.

SUMMARY OF SAFETY ANALYSIS

The safety evaluation concluded that no unreviewed safety question exists and all appropriate design basis calculations were completed.

DCP 92-171

FEEDWATER REGULATING VALVES INTERNALS REPLACEMENT
NORTH ANNA UNIT 1

DESCRIPTION

The existing main feed regulating valves (MFRVs) were supplied by Copes-Vulcan and had a history of maintenance and operational problems. Numerous modifications had been performed to the actuator, valve internals and position controller in an effort to eliminate control instabilities and vibration induced failures of the valves. Those efforts for the most part were not successful. This design change was implemented to replace the internals of the existing valves with a "characterized" pressure reducing trim which would yield a longer valve stroke and the actuators were replaced with stronger piston actuator units which were less susceptible to the effects of flow vibration and pressure differential across the valve.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-005)

The replacement of the MFRVs internals and actuators did not constitute an unreviewed safety question as defined in 10CFR50.59 since it did not:

A) Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in UFSAR.

The new valve trim had a lower flow coefficient than the existing valves when both were full open. Thus, the affects of excessive feedwater addition due to a valve going full open were reduced. The ability of the MFRVs to isolate feedwater when required was not changed. The design was such that fail safe actuation was preserved and Tech Spec response time requirements were maintained. The modification improved the reliability and control of the FW system. The potential for the valve failing open or closed due to a positioner malfunction was not increased since that portion of the valve/actuator was retained from the existing valve/actuator. The actuator and air receiver were seismically qualified. Thus, the potential for loss of Instrument Air due to the modification was not increased.

B) Create a possibility for an accident or malfunction of a different type than any evaluated previously in the UFSAR.

Even though a different type of actuator (air vs spring to close) was used with additional components (AOVs and an air receiver), the possibility of a different type of malfunction was not introduced. The new actuator assembly was seismically qualified and the air tank was seismically installed. In addition the AOVs were spring loaded and designed fail-safe so that air was directed to close the valve. The check valve assembly ensured that a loss of Instrument Air did not prevent the valve from closing.

C) Reduce the margin of safety as defined in the basis of any Technical Specification.

Tech Spec closure times were maintained by the new design. Since FW would still be isolated within the time given in the Tech Specs, the margin of safety was not reduced.

SAFETY EVALUATION NUMBER 93-SE-MOD-005

DESCRIPTION

This SE evaluates the replacement of internals and actuators on the MFRV (1-FW-FCV-1478, 1488, 1498 and 2-FW-FCV-2478, 2488, 2498).

SAFETY EVALUATION SUMMARY

The new actuator differs from the existing actuator in that it uses air to open and close versus air to open and spring to close for the existing actuator. The valve internals and bonnet are designed for use on a 900 lb pressure class valve. Therefore, the integrity of the pressure boundary would be maintained. In addition, the actuator, AOVs, and associated tubing and fittings are designed for use on the MFRVs. The EZD for the Mechanical Equipment Room states that the area is considered a mild environment and the potential for a high energy line break is not postulated due to the extensive inservice inspection program coupled with the leak detection system. The O-rings and diaphragms in the actuator and AOVs are rated for this environment. The potential for failure of the pneumatic actuators to close the MFRVs on an ESF signal is not increased for the following reasons:

- The signals to close the valves have not been changed.
- Each MFRV will have a dedicated air receiver to close the valve on a signal.
- The valve/actuator assembly is seismically qualified and the air receiver and associated tubing will be seismically mounted.
- The FW piping has been analyzed by Engineering Mechanics and it was concluded that the seismic integrity of the line is maintained.
- The pneumatic actuator system is fail safe in that the AOVs reposition on a loss of air to their fail safe position which vents air from below the actuator piston and injects air above the piston.
- The air receiver is continuously kept pressurized by the IA system and a check valve arrangement prevents the receiver from depressurizing on a loss of IA.

Use of the pneumatic actuator will not increase the probability of an accident since it is designed for this application, seismically qualified and fail safe as was the existing actuator. Implementation will reduce the magnitude of an excess FW addition accident since the new trim has a lower flow coefficient when full open but is sized adequately to provide the required flow at 100% power. Even though the pneumatic actuator utilizes additional components, the possibility for a different type of accident or failure is not introduced since these components are qualified for use in this application and fail safe.

SAFETY EVALUATION NUMBER 93-SE-MOD-005 Rev. 1

DESCRIPTION

This SE evaluates the replacement of internals and actuators on the MFRV (1-FW-FCV-1478, 1488, 1498 and 2-FW-FCV-2478, 2488, 2498).

SAFETY EVALUATION SUMMARY

The new actuator differs from the existing actuator in that it uses air to open and close versus air to open and spring to close for the existing actuator. The valve internals and bonnet are designed for use on a 900 lb pressure class valve. Therefore, the integrity of the pressure boundary would be maintained. In addition, the actuator, AOVs, and associated tubing and fittings are designed for use on the MFRVs. The EZD for the Mechanical Equipment Room states that the area is considered a mild environment and the potential for a high energy line break is not postulated due to the extensive inservice inspection program coupled with the leak detection system. The O-rings and diaphragms in the actuator and AOVs are rated for this environment. The potential for failure of the pneumatic actuators to close the MFRVs on an ESF signal is not increased for the following reasons:

- The signals to close the valves have not been changed.
- Each MFRV will have a dedicated air receiver to close the valve on a signal.
- The valve/actuator assembly is seismically qualified and the air receiver and associated tubing will be seismically mounted.
- The FW piping has been analyzed by Engineering Mechanics and it was concluded that the seismic integrity of the line is maintained.
- The pneumatic actuator system is fail safe in that the AOVs reposition on a loss of air to their fail safe position which vents air from below the actuator piston and injects air above the piston.
- The air receiver is continuously kept pressurized by the IA system and a check valve arrangement prevents the receiver from depressurizing on a loss of IA.

Use of the pneumatic actuator will not increase the probability of an accident since it is designed for this application, seismically qualified and fail safe as was the existing actuator. Implementation will reduce the magnitude of an excess FW addition accident since the new trim has a lower flow coefficient when full open but is sized adequately to provide the required flow at 100% power. Even though the pneumatic actuator utilizes additional components, the possibility for a different type of accident or failure is not introduced since these components are qualified for use in this application and fail safe.

SAFETY EVALUATION NUMBER 93-SE-MOD-005 Rev 2

DESCRIPTION

Replace the internals and actuators on the Main Feedwater Reg Valves.

SAFETY EVALUATION SUMMARY

The new actuator will not increase the probability of an accident since it is designed for this application, seismically qualified and fail safe.

DCP 93-109

REPLACE MAIN STEAM TRIP VALVE
RUPTURE DISKS
NORTH ANNA UNIT 1

DESCRIPTION

The main steam trip valve rupture disks were replaced with disks made from inconel. The original disks were susceptible to fatigue if they were operated in a high temperature environment which could cause premature disk failure. With the ambient temperatures experienced in the area of the valves, the disks were being operated above the recommended maximum of 70% of burst pressure. The new disks can be operated at up to 90% of their rated burst pressure and have a cycle life ten times greater than the original disks.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-006)

This design change did not create an unreviewed safety question as defined by 10CFR50.59.

A. The rupture disk replacement did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

The pressure rating of the rupture disks was not changed so that the operation of the main steam trip valves was not affected. Closure rate for the valves upon a closure signal is within the required 5 seconds.

B. The implementation of this modification did not create a possibility for an accident of a different type than previously evaluated in the Final Safety Analysis Report.

The disk replacement was an enhancement to increase disk reliability. The operation of the valves was not changed and system design was not affected so that the possibility of accidents of a different type was not created.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

The trip valves are still required to be operational in modes 1,2 and 3 and close within the required 5 seconds. The margin of safety is not affected.

DCP 93-110

REPLACE MAIN STEAM TRIP VALVE RUPTURE DISKS NORTH ANNA UNIT 2

DESCRIPTION

The main steam trip valve rupture disks were replaced with disks made from inconel. The original disks were susceptible to fatigue if they were operated in a high temperature environment which could cause premature disk failure. With the ambient temperatures experienced in the area of the valves, the disks were being operated above the recommended maximum of 70% of burst pressure. The new disks can be operated at up to 90% of their rated burst pressure and have a cycle life ten times greater than the original disks.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-006)

This design change did not create an unreviewed safety question as defined by 10CFR50.59.

A. The rupture disk replacement did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

The pressure rating of the rupture disks was not changed so that the operation of the main steam trip valves was not affected. Closure rate for the valves upon a closure signal is within the required 5 seconds.

B. The implementation of this modification did not create a possibility for an accident of a different type than previously evaluated in the Final Safety Analysis Report.

The disk replacement was an enhancement to increase disk reliability. The operation of the valves was not changed and system design was not affected so that the possibility of accidents of a different type was not created.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

The trip valves are still required to be operational in modes 1,2 and 3 and close within the required 5 seconds. The margin of safety is not affected.

SAFETY EVALUATION NUMBER 93-SE-MOD-007

DESCRIPTION

Modify the North Anna Setpoint Document by changing the EDG lube oil low temperature alarm to 90°F. The lube oil filter differential pressure will be changed from 15 to 13 psid. Correct the UFSAR regarding the EDG jacket water low pressure setpoint and reset values. Maximum outside air temperature should be 104°F based on a maximum 110°F radiator inlet temperature and 6°F temperature rise across the room. The EDG room fan capacity is 5000 CFM not 50000 CFM.

SAFETY EVALUATION SUMMARY

These changes are consistent with the Colt Industries Service Manual for the North Anna EDGs. No unreviewed safety questions exist.

SAFETY EVALUATION NUMBER 93-SE-MOD-008

DESCRIPTION

Repair/replacement of the Unit 2 SW lines to the Unit 2 QS building.

SAFETY EVALUATION SUMMARY

To repair the headers, the lines will be blanked off from the rest of the system so that the Unit 1 SW system and those portions of the system common to both units will be intact. Exposure of piping embedded in concrete will result in a total CDF increase of less than 1E-6 as long as measures for preventing construction mishaps are implemented. Excavation will not start until after NRC relief from missile protection requirements.

DCP 92-196

REPLACEMENT OF STEAM TRAPS MAIN STEAM VALVE HOUSE NORTH ANNA UNIT 1

DESCRIPTION

Several steam traps in the Main Steam Valve House were defective and needed to be replaced. The steam trap model which was installed when the plant was built is no longer available in the original configuration.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-009)

The defective traps were replaced with either Yarway 515 ASWR or Spirax Sarco TD 120 steam traps. These traps meet all the design requirements for pressure, temperature and flow. The new steam traps operate in the same manner as the original equipment. As some of the replacement traps did not have strainer drain connections, several drain valves were eliminated. Seismic qualification of the steam trap piping was not affected. Failure of the pressure boundary of any of the new steam traps will result in a steam leak equal to or less than that which would result from a failure of a 1" pipe. This would be less than the 6" break previously analyzed in the UFSAR.

SAFETY EVALUATION NUMBER 93-SE-MOD-010

DESCRIPTION

Defeat the auto closure of containment purge and exhaust isolation MOVs on a high-high signal from the containment gaseous/particulate and manipulator crane rad monitors to support maintenance.

SAFETY EVALUATION SUMMARY

Tech Specs and the UFSAR do not require this function with the reactor defueled. No outage activities will be in progress with the function defeated. An action statement will ensure Mode 6 entry is prevented during the tagout.

SAFETY EVALUATION NUMBER 93-SE-MOD-011

DESCRIPTION

Evaluate the installation of 2-CH-PI-2111B "boric acid filter PI" and upgrade the materials to safety related.

SAFETY EVALUATION SUMMARY

The gauge was installed on the unistrut stand per the original plant configuration. The installation of capillary tubing will not affect the operation of the indicator. There will be no impact on malfunctions of equipment important to safety.

DCP 93-118
Abandonment of One In-Core Thermocouple
(1-IC-TE-37)
North Anna Unit 1

Description

One in-core, core exit, thermocouple was abandoned in place because it is inoperable. The thermocouple will not be repaired or replaced due to ALARA concerns, inaccessibility, and the probability of damaging additional thermocouples in the core during repair efforts. The Technical Specification requirement of two operable thermocouples per quadrant per train was not impacted by this modification. Abandonment consisted of removing the computer points, for the thermocouple, from the ERFCS and ICCM computer software.

Summary of Safety Analysis (-SE-MOD-012)

This design change in accordance with DCP 93-118 does not create an "unreviewed safety question" as defined in 10 CFR 50.59.

A. The implementation of this modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

   One in-core, core exit thermocouple was abandoned in place. The remaining thermocouples provide sufficient data for monitoring core exit temperatures. The in-core thermocouples do not provide any control or protection functions. Abandonment does not prevent the Operators from performing necessary measures to mitigate an accident.

B. The implementation of this modification does not create a possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report.

   The in-core thermocouples do not provide any control or protection functions. Abandonment does not prevent the Operators from performing necessary measures to mitigate an accident. No feedback into protective circuitry is possible. This modification removed the associated computer input points from the ICCM and the ERFCS computers. No other computer input is impacted.

Summary of Safety Analysis (continued)

C. The implementation of this modification does not reduce the margin of safety as defined in any Technical Specification.

The minimum required number of operable thermocouples (Section 3.3.3.6) per Technical Specifications is not impacted. At least five operable thermocouples per quadrant per train are available.

SAFETY EVALUATION NUMBER 93-SE-MOD-013

DESCRIPTION

Abandonment of incore thermocouples 2-IC-TE-11 and 2-IC-TE-32.

SAFETY EVALUATION SUMMARY

The operator's ability to monitor core temperature will not be adversely affected by removing the invalid data from the ICCM and the ERF computer. The minimum required thermocouples per quadrant per train will not be exceeded. These inputs do not perform any control or protective functions.

SAFETY EVALUATION NUMBER 93-SE-MOD-014

DESCRIPTION

Replacement of Farris Type 2740 relief valves with 2740-ULR relief valves in the CC system.

SAFETY EVALUATION SUMMARY

The function and operation of the relief valve and the system in which it is installed remains the same.

DCP 93-116

REPLACE VARIOUS SOVs ON THE UNIT 1 FW SYSTEM
(SE #93-SE-MOD-015)

DESCRIPTION

The existing Solenoid Operated Valves (SOVs) (1-FW-SOV-1479-2 and 1-FW-SOV-1489-2) to the FW bypass valves failed and are no longer made. The valves were replaced with a similar SOV made by the same manufacturer (ASCO).

SUMMARY OF SAFETY ANALYSIS

All applicable design calculations were revised and the safety evaluation concluded that an unreviewed safety question does not exist.

SAFETY EVALUATION NUMBER 93-SE-MOD-016

DESCRIPTION

Replace the existing RC-LT-1000 and RC-LT-2000 with a new model.

SAFETY EVALUATION SUMMARY

The five valve manifold arrangement and new transmitter are qualified for the environment they will be exposed to, and the tubing will be connected using approved fittings and welds. Thus, the RCS boundary will not be affected.

DESIGN CHANGE PACKAGE 93-134
REPLACE FLOW TRANSMITTER ON THE RC SYSTEM
NORTH ANNA UNIT 1

DESCRIPTION

Flow transmitter 1-RC-FT-1416 (Loop I Channel III) was replaced, along with its associated manifold. The old instrument arrangement was a Foxborrow E13DH transmitter with a Kerotest manifold. The new arrangement is a Rosemount 1153HD5PA with a 5 valve arrangement. The modification required an additional structural support be added to rack 1-113 to support the 5 valve arrangement.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-017)

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

1. Accident probability has not been increased because the new transmitter and 5-valve manifold arrangement will not increase the chances of occurrence of any of the accidents considered. The transmitter detects low flow conditions in order to mitigate the accident by tripping the reactor and does not affect probability for an accident. The manifold is being installed in accordance with all applicable specifications and the probability for leakage is not changing.

2. Accident consequences are not increased. The new transmitter meets all specifications, requirements, codes and GDC which were required for the original. It will operate in the same manner to ensure that the reactor trips in the event of an accident in order to mitigate an analyzed Design Basis Event.

3. No unique accident probabilities are created. The transmitter replacement and 5-valve manifold arrangement will not affect the operation of the RCS. System design bases are unchanged.

4. Margin of Safety is maintained because the integrity and reliability of the systems, RCS and RPS, that the transmitter and 5-valve manifold arrangement serves is unchanged.

DCP 92-303

REPLACE SCREEN ENCLOSURES AT EMERGENCY
DIESEL GENERATOR (EDG) EXHAUST AREA
(SE #93-SE-MOD-018)

DESCRIPTION

The aluminum frames and screens which prevent birds (and other objects) from entering the EDG exhaust areas deteriorated. The enclosures were replaced with carbon steel framing and screens and reanchored with Hilti Anchor Bolts.

SUMMARY OF SAFETY ANALYSIS

The safety evaluation considered screen opening size, method of attachment, and durability and concluded that an unreviewed safety question does not exist.

DCP 92-240

REPLACE PCV WITH PRESSURE REGULATOR
NORTH ANNA UNIT 1

DESCRIPTION

1-RC-PCV-1473 is located on the nitrogen supply line to the Pressurizer Relief Tank. Difficulty existed in maintaining the PRT pressure using the existing assembly. Therefore, the existing valve and controller were replaced with a Fisher model #95L pressure regulator. Additional pipe supports were also installed on the nitrogen line.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-019)

The replacement of pressure control valve, 1-RC-PCV-1473, with pressure regulator, 1-SI-REG-1002 does not constitute an unreviewed safety question or require a modification to the Technical Specifications. The function and operation of the PRT and the Reactor Coolant System remain the same. The replacement does not increase the probability, consequence or possibility of an accident. The Margin of Safety as set forth in the Tech Specs. is not affected. Accidents of a different type than previously analyzed are not possible.

DCP 93-132

INSTALL PIPE SUPPORTS TO 01-CC-RV-123B AND 01-CC-RV-123C
DISCHARGE LINES 3"-CC-367-151 AND 3"-CC-370-151
NORTH ANNA UNIT 1

DESCRIPTION

Additional supports were installed on lines 3"-CC-367-151 and 3"-CC-370-151 to ensure seismic integrity of the lines. The above lines are the discharge lines from 01-CC-RV-123B and 01-CC-RV-123C respectively and are non-safety with the safety classification boundary at the respective relief valves. However; these lines were required to be seismically restrained to prevent overstress of the safety related Component Cooling piping and protect other safety related equipment in the collapse envelop of the lines during a Design Basis Seismic Event.

SUMMARY OF SAFETY ANALYSIS

This design change in accordance with DCP 93-132 did not create an "unreviewed safety question" as defined in 10 CFR 50.59.

A. The implementation of this modification did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

The discharge lines from Component Cooling relief valves 01-CC-RV-123B and 01-CC-RV-123C and associated supports have been analyzed for structural adequacy under a Design Basis Seismic Event. This precluded failure of the supports and piping under such an event which consequently would prevent damage to other safety related equipment in the vicinity. Thus, this modification did not have the potential to increase the probability or consequences of an accident such as seismic event or malfunction of the Component Cooling System or any other system.

B. The implementation of this modification did not create a possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis report.

The support installation for discharge lines from the Component Cooling Relief Valves 01-CC-RV-123B and 01-CC-RV-123C is a passive modification, which did not impact the operation of the system. Also, the discharge piping is non-safety and the analysis of the piping and the supports ensure structural integrity during a Design Basis Seismic Event. Thus, the modification has no potential to impact any other system or equipment in the vicinity. Therefore, this modification did not create a possibility for any other accident or malfunction and did not jeopardize any equipment, system or procedure required to operate the plant safely and achieve and maintain safe shut down or to prevent the release of radiation for any condition.

C. The implementation of this modification did not reduce the margin of safety as defined in any Technical Specification.

The discharge lines to the Component Cooling Relief valves 01-CC-RV-123B and 01-CC-RV-123C and the associated supports had no impact on the Technical Specifications requirements. The piping and the supports did not perform any safety-related function. However; the lines were supported seismically to preclude failure under a Design Basis Seismic Event; so as, to prevent damage to other safety-related equipment in the vicinity. Thus, the margin of safety as defined in the Technical Specifications was not impacted by this modification. Therefore, the Technical Specifications bases remain unaffected as a result of this modification.

DCP 92-246

REPLACEMENT OF MAIN STEAM NON-RETURN BYPASS VALVES NORTH ANNA UNIT 2

DESCRIPTION

The main steam non-return bypass valves 2-MS-NRV-203A, 2-MS-NRV-203B and 2-MS-NRV-203C were worn and an exact replacement was not available. The three Rockwell-Edwards 3" figure 3668MT valves were replaced with Edwards 3" figure B36268MLT5 valves. The new valves are shorter in length by 1.3 inches, weigh 44 pounds less, have a flow coefficient (Cv) of 100 verses 90, have a pressure rating of 1690# verses 1500#, have a stem travel of 2.18" verses 1.875" and take ~33 seconds verses ~28 seconds to stroke.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-021)

This design change did not create an unreviewed safety question as defined in 10 CRF 50.59. A failure of the pressure boundary was considered and the probability and consequences of this accident will not be increased. All secondary break analysis remain fully bounding.

The small increase in time (~28 to ~33 seconds) it will take to stroke the new valves closed is not significant. These valves are used during start-up to warm the main steam system and to equalize the pressure across the main steam headers non return valves. There are no technical specifications applicable to these NRV bypass valves.

SAFETY EVALUATION NUMBER 93-SE-MOD-023

DESCRIPTION

Conduct seismic walkdowns on safe shutdown equipment and verify seismic adequacy.

SAFETY EVALUATION SUMMARY

Activities to be conducted are passive in nature in that no equipment will be modified and no methods of operation will be changed.

DCP 93-138

REDESIGN OF FEEDWATER MONOBALL SLIDING RESTRAINTS
(SE #93-SE-MOD-024)

DESCRIPTION

During the 1993 Unit 1 Refueling outage it was reported that Feedwater (FW) monoball sliding restraints 1-FW-PH-5 and 1-FW-PH-13 in the Mechanical Equipment Room were cracked in the top collar section. The monoballs were found to be offset within the top restraint plate and tight against one side in the cold position. The cracks were through the collar thickness and extended approximately 2 1/4" from the top of the collar. The DCP replaced these supports with a simpler sliding restraint utilizing existing welded pipe attachments and the existing welded baseplate. The intermediate components were replaced by a tube steel and bearing plate arrangement.

SUMMARY OF SAFETY ANALYSIS

The safety evaluation concluded that an unreviewed safety question did not exist since the new sliding restraint is capable of transferring all design forces and displacements.

SAFETY EVALUATION NUMBER 93-SE-MOD-025

DESCRIPTION

Replacement of Farris Type 2740 relief valves with 2740-ULR relief valves.

SAFETY EVALUATION SUMMARY

The function and operation of the relief valve and the system in which it is installed remains the same.

DCP 93-156

REPLACE BLOWDOWN MANIFOLD FOR
1-FW-LT-1495
NORTH ANNA UNIT 1

DESCRIPTION

The "C" steam generator channel II narrow range level transmitter was equipped with a Hoke two valve manifold on the low side inlet drain line. This manifold was supplied with a test tap between the two valves which was never used by maintenance. Both of the valves needed to be repaired but repair parts and replacement manifolds are no longer available. The two valve manifold was removed and replaced with two Anderson Greenwood needle valves.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-026)

This design change did not create an unreviewed safety question as defined by 10CFR50.59.

A. The replacement of the two valve manifold with two isolation valves did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

The transmitter detects low low level conditions in the SG to mitigate an accident by tripping the reactor and does not affect probability for an accident. The replacement valves meet all specifications, requirements and codes which were required for the original manifold. The valves operate in the same manner, as isolation when the plant is operating, and do not affect the trip function of the transmitter so that accident consequences were not increased.

B. The implementation of this modification did not create a possibility for an accident of a different type than previously evaluated in the Final Safety Analysis Report.

The replacement of the manifold with two isolation valves did not affect the operation of the transmitter or the feedwater system. System design bases are unchanged.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

The integrity and reliability of the feedwater system and reactor protection system, that the transmitter isolation valves serve, is unchanged.

DCP 93-176

REMOVE VALVE 2-CH-321 AND LINE 3/4-CH-934-1502-Q1
(SE #93-SE-MOD-027)

DESCRIPTION

The thermal barrier test line developed a flange leak between the "C" Reactor Coolant Pump and valve 2-CH-321. The line could not be isolated and a repair was needed. The line was originally used during pre-operational testing to set up the seal injection flows but later model Westinghouse Reactor Coolant Pumps eliminated this and similar test connections. This DCP removed the valve, piping, and blind flange up to the leaking flange. A new gasket and blind flange were installed utilizing a bolted connection detail.

SUMMARY OF SAFETY ANALYSIS

The safety evaluation concluded that an unreviewed safety question does not exist since the line and valve are no longer needed and seismic integrity is maintained. Proper installation precautions were noted in the safety evaluation.

DCP 92-168-3

FUEL OIL PIPING UPGRADE
NORTH ANNA UNITS 1 & 2

DESCRIPTION

On 2/26/92, a failure occurred to the 1-AB-P-8A pump suction strainer drain piping connection due to overtorqueing of the drain piping threaded end cap. The design change replaced the fuel oil transfer pumps (FOTP) and aux boiler (AB) pump's suction strainer drain piping with schedule 80 piping in order to provide a more rigid design than the existing schedule 40 piping arrangement.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-073)

The modification to the AB and EDG FOTP suction strainer drain pipe did not constitute an unreviewed safety question as defined in 10CFR50.59 since it did not:

A) Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in UFSAR.

The AB and EDG FOTP suction strainer drain pipe was replaced with a stronger, more rigid schedule 80 pipe. The modification improved system reliability and integrity without affecting system operation. The probability of an accident occurring was not increased and the consequences of previously analyzed accidents remained unaffected.

B) Create a possibility for an accident or malfunction of a different type than any evaluated previously in the UFSAR.

Replacement of the FO piping did not affect the operation of the FO system nor was the integrity of the system adversely affected. The chances of a loss of FO for the EDGs or ABs were not increased

C) Reduce the margin of safety as defined in the basis of any Technical Specification.

Operation of the FO system remained the same following strainer drain pipe replacement and thus, the margin of safety as described in the Tech Specs was not reduced.

DCP 93-006

SERVICE WATER PIPING REPAIR/REPLACEMENT
NORTH ANNA UNIT 2

DESCRIPTION

The SW lines to the Unit 2 Quench Spray Building, buried and embedded in concrete, are deteriorated due to pitting corrosion and require repair/replacement. As the first step of this repair/replacement, the circumferential welds in the high stress areas of these 24" headers were repaired during the 1993 Unit 2 refueling outage. During the repair, these headers were temporarily blanked off the rest of the system which allowed the Unit 1 SW system and those portions of the SW system common to both Units to remain in service. Also the 24" Auxiliary SW discharge lines downstream of the isolation valves from the SW header to the Unit 2 Circulating Water discharge tunnel were repaired/replaced.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-058)

This Design Change did not involve an unreviewed safety question: The project required isolation of one train of service water for brief periods to install and subsequently remove piping plugs to the affected sections. Both or one unit operation at power with a single train of service water is allowed for periods of up to seven days by plant technical specifications. However, reducing the redundancy of the service water system through repeated use of the seven day action statement may contribute to slightly increasing the risk of an accident resulting in core damage. PRA results indicate that the repeated isolation of SW headers for the purpose of installing and removing plug will have a small effect (1.8E-8 increase) on the probability of a CD event for Unit 1.

In the very unlikely event of complete loss of SW during TS Action when Unit 2 was in mode 5 or 6, provisions would have been made to utilize Bearing Cooling water to the Unit 1 CR chillers by placing the crosstie installed under DC 91-009-1 in service.

Repair/replacement of the ASW discharge lines was done within the existing TS, one at a time, when the Unit 2 Circulating Water Tunnel is out of service. In case of events which required restoration of the ASW system, a SIDU plug would have been installed in the ASW line under repair, and the tunnel would have been placed in service for the ASW discharge.

DCP 93-203

REMOVAL OF PIPING AND VALVES ASSOCIATED WITH RCP TEST CONNECTIONS NORTH ANNA UNIT 2

DESCRIPTION

Early in 1993 a leak developed at a flanged connection on a thermal barrier test line on a reactor coolant pump in Unit 2. Design Change 93-176 was written and implemented to repair the leak and eliminate its source by removing a spool piece and installing a blank flange at the pump connection.

To further reduce the probability of leakage from these connections Design Change 93-203 removed the remaining spool pieces, cut the connections and capped them at the Unit 2 reactor coolant pumps. The thermal barrier test connections were used during pre-operational testing to set up the seal injection flow rates. There is no use for these connections in the forseeable future.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-076)

Removing the unused spool pieces and capping the connections will not affect normal or emergency operation of the unit. The pressure boundaries of the Chemical and Volume Control and Reactor Coolant systems are maintained.

An unreviewed safety question does not exist because:

1. The probability of accidents such as loss of coolant or RCS depressurization is slightly reduced by elimination of possible leak sites.

2. Consequences of accidents will not be affected because the test connections are not used to mitigate accident consequences.

3. No unique accident probabilities are created and margin of safety is maintained.

DCP 93-018-2

REACTOR COOLANT PUMP UNDERFREQUENCY CABLE SEPARATION

DESCRIPTION

The 125VDC control power supply cables to the Reactor Coolant Pump (RCP) Reactor Protection System (RPS) Underfrequency (UF) circuits were designated Neutral (Black) and routed accordingly. As a result, Unit 2 DC power cables were routed in common raceways. Routing the cables together introduced the possibility that a single failure, an internally generated short circuit, could result in loss of more than a single RCP UF input to the RPS. The permanent resolution implemented by this Design Change was to color code the power supply cables and to install new cables in the color coded raceway. This ensures that train separation is provided per IEEE-279, 1971 as defined in the Final Safety Analysis Report.

SUMMARY OF SAFETY ANALYSIS: Safety Evaluation # 93-SE-MOD-077

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

A. The implementation of this modification did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Final Safety Analysis Report.

   Neither the probability of occurrence, nor the consequences of a complete loss of forced reactor coolant flow have increased as a result of this DCP. Additionally, there is no increased probability of losing the reactor coolant pumps or the RCP Bus UF relays since there are no modifications to the logic or protective setpoints. All three channels now meet the required separation specification since the implementation of this DCP.

B. The implementation of this modification did not create the possibility for an accident or malfunction of a different type than evaluated previously in the Final Safety Analysis Report.

   This DCP did not create the possibility for an accident or malfunction of a different type than any evaluated previously because it provided the separation required by IEEE-279 which is required by the SAR.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

All setpoints, system descriptions and surveillance requirements as described in the Technical Specifications remain unchanged. There is no reduction in the margin of safety denoted in the Technical Specification Basis Section 2.2.1 - Undervoltage and Underfrequency RCP Busses.

DCP 93-255

REPLACE MISSION DUO CHECK VALVE
NORTH ANNA UNIT 2

DESCRIPTION

The seat on the component cooling water (CCW) containment isolation valve, 2-CC-199, was damaged and could not be repaired. The valve was a Mission Duo Check valve which has a long lead time. The valve was replaced with an Anderson Greenwood CV1B wafer check valve.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-078)

This design change did not create an unreviewed safety question as defined by 10CFR50.59.

A. The replacement of the Mission Duo Check with the Anderson Greenwood CV1B wafer check did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

   The CCW system is used for the orderly shutdown of the unit as long as there is no phase B isolation which occurs after an accident so that the probability of occurrence has not been affected. The CCW to containment can be supplied from either of the CCW headers so that redundancy and the consequences of an accident are not affected.

B. The implementation of this modification did not create a possibility for an accident of a different type than previously evaluated in the Final Safety Analysis Report.

   The design function of the valve has not been changed. The new check valve will still close during a Phase B isolation and remain open during normal operations.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

   The integrity and reliability of the CCW system and the valve was not affected by the valve change as the replacement valve was manufactured to all applicable codes and standards, and installed in accordance with plant specifications.

DCP 93-243

REMOVE LOOP STOP VALVE LOWER TAPS
NORTH ANNA UNIT 2

DESCRIPTION

The original disc pressurization system for the loop stop valves required two valves to pressurize between the discs. A new method of disc pressurization is now being used which only requires one valve and relies on the accumulators to supply pressurization. As the second valves were no longer used they were removed to eliminate potential leakage sites.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-074)

This design change did not create an unreviewed safety question as defined by 10CFR50.59.

A. The removal of the loop stop valve lower disc pressurization valves did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

   The removal and capping of the lines reduced the number of sites for possible RCS leakage. However, leakage of the pipe caps is still bounded by the small break LOCA analysis.

B. The implementation of this modification did not create a possibility for an accident of a different type than previously evaluated in the Final Safety Analysis Report.

   The only function of the valves was as system pressure boundaries. The caps serve the same safety function with less possible leakage sites.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

   The integrity and reliability of the RCS system and the loop stop valves was not affected.

DESIGN CHANGE PACKAGE 93-115
CONNECTION OF RCP FIRE DETECTION PANELS TO
ROBERTSHAW FIRE ALARM PANEL
NORTH ANNA UNIT 2

DESCRIPTION

Surveillance of the RCP heat detector circuits for UFSAR Section 16.2.1.1.1 was noted as being missed in DR N-92-1940 and Special Report N92-714. The required periodicity was 31 days and the current schedule is every outage. The action plan for the Special Report identified several alternatives being evaluated to determine the recommended corrective actions for this problem. Until the discrepancy is resolved, a UFSAR identified alternative of monitoring RCP pump temperatures is required. This modification changes the Unit 2 RCP heat detection annunciation circuits from un-supervised to supervised which changes the testing periodicity to 6 months and provides access for testing from outside containment. The modification involves connecting the RCP heat detection circuits to the Robertshaw panel, and disconnecting them from the fire panel.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-028)

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

The changes to the RCP annunciation circuits do not increase the probability for a fire to occur. Consequences from a fire are not changed since these changes do not degrade the fire protection plan and ensure continued compliance with Appendix R and Appendix A to BTP 9.5.1. Consequence are still bounded by the Appendix R analysis. No accident of a different type are created since fires are still bounded by the original assumptions for Appendix R and Appendix A (single fire criteria). Conduit will be seismically installed to prevent damage to adjacent equipment in a seismic event.

DCP-93-178
REMOVAL OF INITIATE MANUAL NDT PROTECTION ALARM NORTH ANNA UNIT 2

DESCRIPTION

Pursuant to implementation of Technical Specification Change Package No. 262, the Unit 2 "Initiate Manual NDT Protection" alarm should be eliminated. This alarm is no longer required and was removed as new RCS pressure setpoints have been established based upon the heat-up and cool-down curves approved under Technical Specification Change Package No. 262. Prior to approval of this Technical Specification Change, the "Initiate Manual NDT Protection "alarm was used as an indication that the Reactor Coolant System (RCS) was approaching a condition which could challenge the integrity of the reactor vessel.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-029)

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

A. The implementation of this modification did not increase the probability of occurrence or consequences of an accident or malfunctions of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

The Manual NDT Protection alarms are a remanent of the NDT protection philosophy in effect under the previous North Anna Unit #2 Technical Specifications. Under this philosophy, an operator response time interval was ensured through adherence to a Technical Specification minimum pressurizer steam volume. In the event of a design basis LTOPS event, the maintenance of a pressurizer bubble ensured that operators would have sufficient time to shut off the charging pump, or to manually actuate the pressurizer PORVs. The Manual NDT Protection required alarms were provided as an indication that the RCS was approaching a condition which could challenge the integrity of the reactor vessel.

B. The implementation of this modification did not create a possibility for an accident or a malfunction of a different type than any previously evaluated in the Final Safety Analysis Report.

Removal of the alarm logic has not impacted any other control logic or actuations. The ability to mitigate overpressure conditions has been maintained.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

Ability of the operators to mitigate overpressure conditions is maintained. Removal of the Manual NDT Protection Required alarm has been removed as part of the implementation of Technical Specification Change Package No. 262 (NRC License Amendment No. 149).

DCP 92-296

ADD TEST SWITCHES TO
UNDERVOLTAGE RELAYS

DESCRIPTION

The Train A and B undervoltage relay circuits for the Reactor Coolant Pump 4KV buses were modified by adding Westinghouse "Flexitest," Type FT-1, test switches. Although it was possible to perform the monthly functional verification on the undervoltage relays required by Technical Specifications, this physical arrangement did not facilitate any adjustments that may be required as a result of that testing. Undervoltage testing on Unit 2 RCP Buses involved work on GE NGV relays in Cabinets 2-EP-CB-28E/F. The NGV relay is not constructed to allow use of a test connection, and it was necessary to lift the wires from the coil and contact terminal points in order to perform the tests. Connections were made with alligator clips. Also, the DC circuits would be tagged and deenergized. The control circuit wire was seven strand and not intended to be routinely manipulated. Westinghouse "Flexitest" Type FT-1 test switches were added to facilitate the testing of these circuits and minimize the consequences of human error. These switches were added to Auxiliary Relay cabinets 2-EP-CB-28E/F in the Instrument Rack Room during the refueling outage for Unit 2 using usual and customary construction and testing methods in accordance with approved station procedures.

SUMMARY OF SAFETY ANALYSIS: Safety Evaluation # 93-SE-MOD-030

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

A. The implementation of this modification did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Final Safety Analysis Report.

The modification added test switches to the undervoltage circuits which are closed during normal unit operation, and they do not change the functional operation of the undervoltage circuits. This modification improves the safety of handling these circuits both during Unit operation and refuelling outages. These circuits perform their Safety Related functions as they did before the modification.

B. The implementation of this modification did not create the possibility for an accident or malfunction of a different type than evaluated previously in the Final Safety Analysis Report.

The addition of these passive test switches did not change the operation or response of the undervoltage circuits. They operate to perform their functions as before the modification, and since they are passive devices they do not introduce any new accident or malfunction that is not already bounded by single failure or common mode analysis.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

The control, function and operating conditions of the undervoltage circuits did not be change with respect to their designed safety related functions by the addition of this minor control circuit improvement. The modification in no way affects the availability of the systems for their safe shutdown functions.

DCP 92-257-2

ADD TEST JACKS TO UNDERFREQUENCY RELAYS

DESCRIPTION

The underfrequency relay circuits for the Reactor Coolant Pump 4KV buses were modified by adding H. H. Smith binding posts to the test circuit. Although it was possible to perform the monthly functional verification on the underfrequency relay equipment it was not constructed to facilitate this testing required by Technical Specifications. Underfrequency testing on Unit 2 RCP Buses involved opening junction box 2-EP-CB-28UD and connecting jumpers to terminal blocks TA-1 and TA-2 in order to provide a variable frequency generator input for the test. Testing was enhanced by the addition of "Banana Plug" test jacks (binding posts) on the exterior of the junction box. The binding posts were installed during the refueling outage for Unit 2 using usual and customary construction and testing methods in accordance with approved station procedures. It is no longer necessary to open the junction box or use alligator clips. Contact with energized circuits and human factor errors are minimized by this design change.

SUMMARY OF SAFETY ANALYSIS: Safety Evaluation # 93-SE-MOD-030

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

A. The implementation of this modification did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Final Safety Analysis Report.

   The modification added test jacks to the underfrequency test circuits which are deenergized during normal operation, and they did not change the functional operation of the underfrequency circuits. This modification improved the safety of handling these circuits both during Unit operation and refuelling outages. These circuits have been performing their Safety Related functions as they did before the modification.

B. The implementation of this modification did not create the possibility for an accident or malfunction of a different type than evaluated previously in the Final Safety Analysis Report.

The addition of these passive test jacks did not change the operation or response of the underfrequency circuits. They operate to perform their functions as before the modification, and since they are passive devices they do not introduce any new accident or malfunction that is not already bounded by single failure or common mode analysis.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

The control, function and operating conditions of the underfrequency circuits was not changed by the addition of this minor control circuit improvement. The modification in no way affects the availability of the systems for their safe shutdown functions.

REPLACEMENT OF P-250 COMPUTER INVERTER
NORTH ANNA UNIT 2

DESCRIPTION

The existing 9 KVA inverter (02-EP-INV-2) located in the Cable Spread Room (CSR) of the Service Building failed and was replaced with a new 10 KVA inverter. The DC power supply for the new inverter was revised. The existing 9 KVA inverter was fed by Station Battery 2-IV and the new 10 KVA inverter is being fed by Station Battery 2-III.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-031)

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

A. The implementation of this modification did not increase the probability of occurrence or consequences of an accident or malfunctions of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

Accident probability has not been increased because the change conforms to standards and admins. The P-250 computer is not utilized for a safety related function, but utilized to assist the operator in the efficient operation of the plant. The computers primary function is to provide the operator with additional information as to the condition of the nuclear steam supply system. It also has the capability to monitor inputs from the balance of plant systems and to alarm and log various off-normal conditions. There is no direct reactor control or protection action taken by the computer; therefore, the safety of the plant operation is not impaired by its loss. The work performed on the Safety Related distribution panels (02-EP-CB-12C & 02-EP-CB-12D) only consisted of relocating a breaker and cabling from 02-EP-CB-12D to 02-EP-CB-12C. All work performed in these panels was controlled by approved Station procedures. Thus, the safety of the plant has not been affected.

B. The implementation of this modification did not create a possibility for an accident or a malfunction of a different type than any previously evaluated in the Final Safety Analysis Report.

The implementation of this DCP was performed during an outage and the operation of P-250 computer system is not required to monitor the conditions of the reactor core. Consequences have not been increased because a failure of the DCP will not corrupt mitigating systems.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

The integrity and reliability of the new inverter has not been affected. In addition, the P-250 computer system and distribution panels are not mentioned in the Tech Specs. This DCP was implemented during an outage.

DCP 92-357

UNIT 2 MAIN STEAM DRAIN LINE MODIFICATIONS
(SE #93-SE-MOD-032)

DESCRIPTION

Additional pipe supports were added to the Unit 2 Main Steam drain line header, 3"-SHPD-405-601-Q3 and 3"-SHPD-420-601 so that long term code allowable stress limits could be met. The deviating condition was discovered after a leak developed on the Unit 1 Main Steam drain line header.

SUMMARY OF SAFETY ANALYSIS

The analysis demonstrated that the as found condition met short term stress limits but that long term stress limits were exceeded in the seismic load case. The safety evaluation concluded that an unreviewed safety question did not exist.

DCP 93-137-2

EDG LOAD SEQUENCING
TIMER REPLACEMENT

DESCRIPTION

Certain Agastat 2400/7000 Series timers used in the Unit 2 Emergency Bus Load Sequencing circuits required replacement. These timers were exhibiting setpoint drift which results from the timer's poor repeat accuracy (+/- 10% of setpoint for times greater than or equal to 200 seconds and +/- 5% for times less than 200 seconds). Several Deviation Reports and Licensing Event Reports were written to document the failures. Six timers selected for replacement were chosen based on the number of failures and the priority of the timer. The Agastat timers were replaced with Allen-Bradley Type RTC having repeat accuracy of approximately +/- 1% which is well within Technical Specification requirements.

SUMMARY OF SAFETY ANALYSIS: Safety Evaluation # 93-SE-MOD-033

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

A. The implementation of this modification did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Final Safety Analysis Report.

   The new timers are performing the same function as the existing equipment. All design parameters have been met or exceeded by the new timers. No modifications to any of the existing input or resultant logic for the timer circuits were altered by this modification.

B. The implementation of this modification did not create the possibility for an accident or malfunction of a different type than evaluated previously in the Final Safety Analysis Report.

   Failure of these timers was bounded by single failure criteria. Additionally, manual operator actions to start the associated equipment was not impacted by this modification. All accidents where a loss of off-site power was postulated or an actuation of the ESF functions were considered in this review.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

Replacing the existing timers with Allen-Bradley timers ensured that the requirements for emergency bus load sequencing has been maintained. The new timers perform the same functions as the original timers.

SAFETY EVALUATION NUMBER 93-SE-MOD-34

DESCRIPTION

DCP-91-009-1 rev 28, installation of caps on non-operating S/G blowdown lines.

SAFETY EVALUATION SUMMARY

This change was the temporary installatio of caps on s/g/ blowdown lines outside the aux building during the SW replace ' repair project. The caps prevents water from dripping into the pipe tunnel betwee. e turbine building and the aux building while this project is underway. Since this piping is not safety related and has been abandoned, this change is allowed.

SAFETY EVALUATION NUMBER 93-SE-MOD-35

DESCRIPTION

DCP-92-301-1 and 2, Various emergency bus breaker relay changes / setpoint changes.

SAFETY EVALUATION SUMMARY

The changes made will enhance the operation of the emergency bus and are being done as a result of an NRC EDSFI. All work will be performed within the bounds of the technical specifications. Overall system design and operation is unchanged.

DCP-92-301-2
REPLACE TRIP DEVICES FOR VARIOUS 4KV EQUIPMENT
NORTH ANNA POWER STATION
UNIT 2

DESCRIPTION

During the July 1991 NRC Electrical Distribution System Functional Inspection, several concerns were identified relating to breaker mis-coordination on safety related switchgear (Reference Finding 91-17-03). Specifically, the 50/51V (voltage restraint overcurrent) relays on the 4KV circuit breakers, each supplying two 4160/480V load center transformers, did not coordinate with the 480V load center breakers associated with the Inside Recirculation Spray Pump (IRSP) and Quench Spray Pump (QSP) motors. The mis-coordination could have resulted in the loss of both transformers and connected 480V load centers for a failure of an IRSP/QSP motor or associated 480V cable(s). Coordination problems were also identified between the ITE trip devices and the IRSP motor supply breakers.

To correct the mis-coordination problems on Unit 2, the 50/51V relays associated with breakers 25H8 and 25J8 needed to have the existing settings for the "taps", "time-dial", an d"instantaneous amps" reset. (These breakers only affect the IRSP and QSP motors). In addition to resetting the relays, several overcurrent devices for the IRSP motors need to be replaced (breakers 24H1-2, and 24J1-2).

The overcurrent trip devices (OD) were replaced on two (2) 480V circuit breakers. These are the supply breakers for the IRSP motors. The "taps", "time-dial" and "instantaneous amps" on the 50/51V relays for breakers 25H8 and 25J8 were reset.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-036)

This design change modified the setpoints for the 4KV transformer feeder breakers 25H8 and 25J8 and the breaker 24H1-2 and 24J1-2 IRSP motor overcurrent trip devices to correct a mis-coordination of overcurrent devices. The revised setpoints continue to meet all design requirements in accordance with applicable industry standards. The needed setpoint changes could not be accomplished using the existing trip devices, therefore, they were replaced.

This setpoint change represents an enhancement and did not constitute an unreviewed safety question as defined by 10CFR50.59.

DESIGN CHANGE PACKAGE 93-105
ADDITION OF P-4 TURBINE TRIP TEST POINTS
NORTH ANNA UNIT 2

DESCRIPTION

Two new terminal blocks, equipped with banana jack adapters, were installed in the reactor trip switchgear cabinet 2-EI-CB-46A. These terminal blocks were wired to points located in the back of the cabinet, which were used by station electricians to determine breaker position during the monthly PT on the SSPS. The new blocks which are located at the front of the cabinet allow easier access to the test points.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-037)

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

A. The implementation of this modification did not increase the probability of occurrence or consequences of an accident or malfunctions of equipment important to safety as previously evaluated in the UFSAR.

   Accident probability has not been increased because this design change conformed to standards and admins. The terminal block installation is an electrical extension of an existing circuit. The installation will not impair or degrade any reactor trip switchgear. This DCP decreases the likelihood of causing an inadvertent unit trip, and does not impair any safety system.

B. The implementation of this modification did not create a possibility for an accident or a malfunction of a different type than any previously evaluated in the UFSAR because the design change is minor and will not affect the operation of the Reactor Trip System.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification because neither of the logic trains will be impaired by this modification.

This activity does not change the basis section of the Technical Specifications and will not create the possibility of an accident of a different type than was previously evaluated in the UFSAR.

This change does not negatively impact operation of a safety-related system or components and does not prevent systems from performing accident mitigating functions.

SAFETY EVALUATION NUMBER 93-SE-MOD-38

DESCRIPTION

DCP-92-272-2 and 273-1, replacement of motors for SW to RSHX supply and return valves.

SAFETY EVALUATION SUMMARY

The new motors will enhance overall system response during and accident and do not adversely affect emergency bus loading. All work will be done within the confines of existing tech specs.

DCP 92-187

Replace PCV with Pressure Regulator
NAPS - Unit 2

DESCRIPTION

02-RC-PCV-2473 was installed in the nitrogen supply line to the Pressurizer Relief Tank. Difficulty existed in maintaining the PRT pressure using this assembly. Therefore, this control valve and associated controller was replaced with a Fisher model #95L pressure regulator.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-039)

The regulator installed meets the design requirements of the nitrogen supply system. The seismic integrity of the supply piping is maintained. The regulator is capable of supplying the necessary amount of nitrogen to maintain the PRT within current operating limits. Since the function and operation of the nitrogen supply system was not changed by this modification, this modification did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the UFSAR, create a possibility for an accident or malfunction of a different type than previously evaluated, or reduce the margin of safety as defined in the Tech. Specs. Therefore an unreviewed safety question does not exist.

DC 92-339

Motor Operated Valve (MOV) Thermal Overload Replacement
North Anna / Unit 2

Description

In their Generic Letter 89-10, the USNRC identified several areas of concern regareding the operability of MOVs. As a result, the utilities were required to prepare and implement a program to improve the operability of safety related MOVs. As a part of the Virginia Power response to the USNRC GL 89-10, the sizes of Thermal Overload elements for all safety related MOVs have been evaluated via calculationEE-0506. Proper sizing of an MOV TOL provides adequate protection of the motor and at the same time assures operability of the MOV under design basis events. The purpose of this DCP was to implement the TOL replacement for the Unit 2 MOVs which was accomplished during the Unit 2 outage.

Summary Of Safety Analysis(93-SE-MOD-040)

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because the MOVs involved are used to respond to an accident which has already occurred.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because the thermal overloads are not called upon to operate unless the MOV has already received a signal to change position and has failed to accomplish that change.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because the MOV operation is not altered and the overall reliability of the MOVs has been increased.

DC 92-272

REPLACE 2#' MOTOR WITH 5#' MOTOR
North Anna / Unit 2

Description

In their Generic Letter 89-10, the USNRC identified several areas of concern regarding the operability of MOVs. As a result, the utilities were required to prepare and implement a program to improve the operability of safety related MOVs. As a part of the Virginia Power response to the USNRC GL 89-10, calculations have been performed for various MOVs. One of these (ME-0317) for the 02-SW-MOV-203A,B,C,D, 02-SW-MOV-204A,B,C,D indicated that the motor size needed to be increased for the valve operator. As a result, the 2#' motors on 02-SW-MOV-203A,B,C,D, 02-SW-MOV-204A,B,C,D were replaced with qualified 5#' motors by this DCP. Also, the motor thermal overloads were replaced with new ones based on calculations which complies with GL 89-10 and STD-GN-0002.

Summary Of Safety Analysis(93-SE-MOD-040)

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because the new components provide margin for delivery of all important design features under all postulated conditions.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because the design remains a like for like replacement in conjunction with standard reviews for GDC 17 and EQ. UFSAR single failure criteria still bounds the design.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because the design function of the MOVs remain the same with stroke time unchanged. The margin of safety is preserved.

DC 92-017

BATTERY 2-BY-B-04 (2-IV)
North Anna / Unit 2

Description

During the 1992 Unit 2 refueling outage, station battery 2-BY-B-04 (2-IV) was capacity tested in accordance with station Periodic Test procedures. The test was terminated at less than full duration by the system engineer performing the test due to the weakness of several cells. Based on the duration completed, the battery is reported to be at 82% of rated capacity, which exceeds the minimum operable criteria per IEEE-450-1987 of 80% of rated capacity. Based on the age of the battery and the test results, the battery was replaced during the 1993 refueling outage. This was required to support continued plant operation in accordance with Technical Specification 3.8.

Summary Of Safety Analysis(93-SE-MOD-040)

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR.

   The replacement of the battery did not increase the probability of any accident as the batteries normally operate in a float mode and hydrogen generated by this and other charging of the battery is dissipated by ventilation. In a LOOP, normally all emergency power systems function properly and both trains of equipment function. In accordance with single failure criteria, one train of emergency equipment can fail to function and an accident still be resolved. Accordingly, this replacement does not increase the consequences of the accidents or of a single failure, if the failure is this battery. This battery is a near one for one replacement of the existing, and is qualified for its service, therefore, its installation did not create a new accident.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR.

Failure of an entire dc and ac vital bus due to failure of the battery has been considered. The impact of such a failure has been considered in the design of the plant and its systems. Due to similarity between the old and new battery, no new types of malfunctions are anticipated. Also, equipment related to the battery which is not being replaced in conjunction with this replacement, have been reviewed and found to be operating within their ratings after this modification is performed. Since the new battery is near full capability and the battery it replaces is old and operating at reduced capacity, installation of the new battery reduces the probability of the battery failing to perform as required for the two hour LOOP duration.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification.

This installation is in accordance with safety related standards and was verified acceptable by testing in accordance with Technical Specification 3.8.2 prior to return to service, thereby assuring that the margin of safety is not reduced. The implementation of this installation was scheduled such that Technical Specification 3.8.2 was not violated with respect to batteries in service. The other station batteries are Exide Model 2GN-23 and are presently performing as required, therefore a change to the Technical Specifications is not required.

DCP 93-145

STATION REROOFING PROJECT
(SE #93-SE-MOD-041)

DESCRIPTION

Various station roofs had deteriorated to the point of needing replacement. DCP 93-145 reroofed the following plant buildings with a single ply membrane roofing system.

LEOF
U1 Auxiliary Feedwater Pumphouse
U2 Auliliary Feedwater Pumphouse
Clarifier Building lower roof
Decon Building lower roof
Vacuum Priming Pumphouse
Crane Enclosure
U2 Rod Drive Room

Appropriate design details were utilized so that the single ply membrane system would give superior service.

SUMMARY OF SAFETY ANALYSIS

The safety evaluation concluded that an unreviewed safety question did not exist. Appropriate precautions were also considered during construction so that a transient would not be caused during installation.

SAFETY EVALUATION NUMBER 93-SE-MOD-42

DESCRIPTION

DCP-90-3-3, installation of security lighting.

SAFETY EVALUATION SUMMARY

This light pole addition does not affect safety related equipment and is needed for security reasons due to the upcoming installation of the station blackout EDG.

SAFETY EVALUATION NUMBER 93-SE-MOD-43

DESCRIPTION

DCP-92-009-3, reclassification of letdown piping.

SAFETY EVALUATION SUMMARY

This is an administrative change to the Q1/Q2 classification of the letdown piping. No physical change to the operation of the CVCS system has been performed nor has the operability of the system been altered.

DCP 92-018

THERMOLAG FIRE BARRIER REPLACEMENT
(SE # 93-SE-MOD-046)

DESCRIPTION

NRC Bulletin 92-01 addressed concerns with the 1 and 3 hour fire rating of Thermo-Lag 330 Fire Barrier Systems. North Anna used the Thermolag 330 fire barrier systems in several 1 and 3 hour fire barrier assemblies. Compensatory measures were taken until the Thermolag was replaced. The power cables for the Unit 1 "C" Charging Pump and Unit 2 "A" Component Cooling Pump were wrapped with 3M Interam E-50 Series Mat. The Thermolag fire barrier in the southeast corner of the 2-CH-P-1A cubicle was replaced with W/R Type C gypsum board.

SUMMARY OF SAFETY ANALYSIS

The safety evaluation considered that compensatory measures would be in place until the modification was completed. The safety evaluation considered all locations of Thermolag and identified locations requiring replacement. The safety evaluation concluded that an unreviewed safety question does not exist.

PHASE A ISOLATION ANNUNCIATOR RECONFIGURATION
NORTH ANNA UNIT 2

DESCRIPTION

The Phase A Isolation annunciator (2K-H7) was reconfigured such that an automatic ESF SI signal from the SSPS or if the "Phase A Isolation" benchboard switches (2CIPAA1 or 2CIPAA2) are held in the "INITIATE" position will activate the annunciator. This DCP performed work in control room benchboards 2-1 and 2-2 and the Solid State Protection System (SSPS) output relay cabinets (02-EI-CB-47E/47F) located in Emergency Switchgear.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-047)

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

A. The implementation of this modification did not increase the probability of occurrence or consequences of an accident or malfunctions of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

Accident probability has not been increased because this design change conformed to standards and admins. The operation of the containment isolation system was not affected. The Engineered Safety Features (ESF) of the SSPS have remained the same. This design change provides a more accurate representation of the status of a Phase A isolation.

B. The implementation of this modification did not create a possibility for an accident or a malfunction of a different type than any previously evaluated in the Final Safety Analysis Report.

The implementation of this DCP was performed during an outage. The operation of the SSPS, SI, Containment Isolation, and Hathaway systems was not affected due to the implementation of this design change. Consequences have not been increased because a possible failure of the DCP would not have corrupted any mitigating systems.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

The integrity and reliability of the Containment Isolation system and ESF have not been affected.

SAFETY EVALUATION NUMBER 93-SE-MOD-48

DESCRIPTION

DCP-92-267-3, structural reinforcement of block wall SB-254-4.

SAFETY EVALUATION SUMMARY

This activity returns the wall to its original design and is a repair.

DCP 93-103

FABRICATE INSTRUMENT MANIFOLD
1-SI-FT-1940-1
NORTH ANNA UNIT 1

DESCRIPTION

The high head safety injection flow transmitter, 1-SI-FT-1940-1, was equipped with a Hoke five valve manifold. The manifold needed to be replaced and an exact replacement was not available. The manifold was replaced with a five valve manifold arrangement, fabricated using Whitey instrument valves.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-049)

This design change did not create an unreviewed safety question as defined by 10CFR50.59.

A. The replacement of the five valve manifold with a five valve manifold arrangement did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

The transmitter is for indication of safety injection flow during post accident monitoring to verify proper SI system operation and does not affect probability for an accident. The replacement valves meet all specifications, requirements and codes which were required for the original manifold. The valves operate in the same manner as the valves in the original manifold and do not affect the function of the transmitter so that accident consequences were not increased.

B. The implementation of this modification did not create a possibility for an accident of a different type than previously evaluated in the Final Safety Analysis Report.

The replacement of the manifold with a fabricated five valve arrangement did not affect the operation of the transmitter or the safety injection system. System design bases are unchanged.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

The integrity, reliability and operation of the safety injection system has not been changed. The required safety injection flow will be available for an accident and will be indicated in the control room.

DCP 93-104

FABRICATE INSTRUMENT MANIFOLD
2-SI-FT-2940
NORTH ANNA UNIT 2

DESCRIPTION

The high head safety injection flow transmitter, 2-SI-FT-2940, was equipped with a Hoke five valve manifold. The manifold needed to be replaced and an exact replacement was not available. The manifold was replaced with a five valve manifold arrangement, fabricated using Whitey instrument valves.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-049)

This design change did not create an unreviewed safety question as defined by 10CFR50.59.

A. The replacement of the five valve manifold with a five valve manifold arrangement did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

   The transmitter is for indication of safety injection flow during post accident monitoring to verify proper SI system operation and does not affect probability for an accident. The replacement valves meet all specifications, requirements and codes which were required for the original manifold. The valves operate in the same manner as the valves in the original manifold and do not affect the function of the transmitter so that accident consequences were not increased.

B. The implementation of this modification did not create a possibility for an accident of a different type than previously evaluated in the Final Safety Analysis Report.

   The replacement of the manifold with a fabricated five valve arrangement did not affect the operation of the transmitter or the safety injection system. System design bases are unchanged.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

   The integrity, reliability and operation of the safety injection system has not been changed. The required safety injection flow will be available for an accident and will be indicated in the control room.

DCP 93-141

RHR PUMP RELIEF VALVE SPRING REPLACEMENT
NORTH ANNA UNIT 2

DESCRIPTION

RHR pump suction relief valves (RVs) 2-RH-RV-2721A&B were designed to provide overpressure protection of the RHR system in the event of leakby from the reactor coolant system. The original design of the RHR system had only one RV installed on the discharge header from the pumps. This RV was to be set to 600 psig. In 1977 Stone & Webster Engineering made a modification to remove the single relief from the discharge of the pump and to install one relief per pump on the suction side. A lift setpoint of 450 psig was originally selected and the springs were changed to accommodate the new setpoint. However the setpoint was later changed to 467 psig. The nameplate data was never changed to annotate the new setpoint.

It was discovered that the installed springs had a set range of 411 to 450 psig. This design change was implemented to replace the existing springs with springs which had a setting range of 451 to 492 psig. The RV lift setpoint of 467 psig would then be encompassed by the design setting range of the spring.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-051)

The replacement of the relief valve spring did not constitute an unreviewed safety question as defined in 10CFR50.59 since it did not:

A) Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in UFSAR.

The RHR pump suction relief valve springs were replaced to ensure that the lift setpoint was encompassed by the design setting range of the new springs. The setpoint was not affected and operation of the RVs was not changed. The RHR system remains isolated from the RCS during power operations and is not taken credit for in mitigation of any accidents.

B) Create a possibility for an accident or malfunction of a different type than any evaluated previously in the UFSAR.

The Design Change increased the reliability of the RHR system. Operation and integrity of the RVs was not affected. The chances of overpressurization had not been increased and the failure of a RHR line at power would be bounded by the LOCA analysis.

C) Reduce the margin of safety as defined in the basis of any Technical Specification.

Implementation of the Design Change improved the reliability of the RHR system and overpressurization protection for the piping was maintained. No margin of safety was reduced or impacted for the basis section of the Technical Specifications.

DCP 93-133

INSTALLATION OF REPLACEMENT PIEZOMETERS
(SE #93-SE-MOD-052)

DESCRIPTION

Various pneumatic piezometers had failed at the Main Dam and the Service Water Reservoir. The piezometers approached or exceeded their design life of 20 years. The piezometers are used for groundwater monitoring of the phreatic surface at the Main Dam and the Service Water Reservoir in accordance with Reg Guide requirements. The pneumatic piezometers were replaced with open tube piezometers.

SUMMARY OF SAFETY ANALYSIS

The safety evaluation concluded that an unreviewed safety question does not exist since compliance with Reg Guide 1.127 and the North Anna Technical Specification is maintained. The new type of piezometer is an acceptable replacement.

SAFETY EVALUATION NUMBER 93-SE-MOD-44

DESCRIPTION

DCP-92-363 and 93-185, replacement of the EDG air start tank pressure indicator.

SAFETY EVALUATION SUMMARY

The replacement indicators have a narrower scale and are therefore easier to read when looking for small pressure changes. The operation and design of the EDG air start system is unchanged.

SAFETY EVALUATION NUMBER 93-SE-MOD-45

DESCRIPTION

DCP-92-18-3, modification of charging pump cable supports due to increased loading.

SAFETY EVALUATION SUMMARY

Fire suppression of the cables was improved which necessitated the strengthening of the cable supports. The operation of the charging system and the electrical distribution system is unchanged.

SAFETY EVALUATION NUMBER 93-SE-MOD-50

DESCRIPTION

DCP-93-175-3, replacement of temperaturs switches for the rod drive room air supply fans.

SAFETY EVALUATION SUMMARY

This is a "like kind" replacement with the only difference is the increased temperature range of the new switches so that the fans will not have to be maintained in the OFF position to prevent from running continuously. This DCP restores original design capability to the rod drive room air supply fans.

SAFETY EVALUATION NUMBER 93-SE-MOD-53

DESCRIPTION

DCP-93-277-3, install wire handrail in EDG exhaust houses.

SAFETY EVALUATION SUMMARY

This DCP was determined not to affect the integrity of the concrete structure or the operability of any of the EDG components.

SAFETY EVALUATION NUMBER 93-SE-MOD-54

DESCRIPTION

DCP-92-17-2, replacement of station battery 2-IV.

SAFETY EVALUATION SUMMARY

The replacement battery is larger than its original and meets all required design criteria. The replacement will be performed while the unit is in modes 5 or 6 IAW existing Tech Specs.

SAFETY EVALUATION NUMBER 93-SE-MOD-55

DESCRIPTION

EWR 92-167, setpoint change for EDG fuel oil day tank high level pump trip.

SAFETY EVALUATION SUMMARY

This setpoint change corrects a problem with inadvertent fuel oil pump tripping due to sensed high level because of the low pressure in the EDG room during EDG runs. All function of the day tank level system still function as designed.

SAFETY EVALUATION NUMBER 93-SE-MOD-56

DESCRIPTION

DCP-93-161-1, Installation of new RC bypass flow controller manifolds.

SAFETY EVALUATION SUMMARY

This is a "like kind" replacement of RCS pressure boundary piping.

SAFETY EVALUATION NUMBER 93-SE-MOD-57 (rev 0 and 1)

DESCRIPTION

DCP 91-14-3, installation of new plant radio system (850 MHz).

SAFETY EVALUATION SUMMARY

Special Test 1-ST-103 (see separate 50.59 review) was completed which proved no adverse plant affects from radio frequency interference caused by the new system. Installation involved control room pressure boundary breaching, which was done IAW existing tech specs.

SAFETY EVALUATION NUMBER 93-SE-MOD-059

DESCRIPTION

DCP 93-011-3 was written to repair/replace 24" aux SW piping.

SAFETY EVALUATION SUMMARY

DCP was written by engineering to give direction on repair/replacement of Aux. SW piping embedded in concrete. Portions of these lines are deteriorated due to general pitting corrosion. The SE determined that there were no unreviewed safety questions.

SAFETY EVALUATION NUMBER 93-SE-MOD-061

DESCRIPTION

DCP 93-102 was written by engineering to reorient RV's on the CC lines.

SAFETY EVALUATION SUMMARY

Several component cooling water relief valves are to be reoriented so that they are installed vertically. These valves are the boron evaporator distillate cooler relief valves, 01-CC-RV-109A/B, the reactor coolant cold leg sample cooler relief valves, 01-CC-RV-113, and the PZR liquid space sample cooler relief valves, 01-CC-RV-111,211. The SE determined that there are no unreviewed safety questions.

SAFETY EVALUATION NUMBER 93-SE-MOD-062

DESCRIPTION

DCP 93-004-2 was written to add bonnet vents on SI system valves.

SAFETY EVALUATION SUMMARY

The DCP was written by engineering to address the problems encountered with LHSI pump PT's. Each time the LHSI pump PT is performed the RV's on the lines start weeping. The vents will allow venting of air from the SI lines to help keep the lines filled with water. This will reduce the pressure spike on the system each time a pump was started. The SE determined that there were no unreviewed safety questions.

SAFETY EVALUATION NUMBER 93-SE-MOD-063

DESCRIPTION

DCP 93- 181-3 was written to install a cable/cord anchoring device.

SAFETY EVALUATION SUMMARY

The DCP was written by engineering to address cable/cord anchoring devices protected by SEMKIT seal passing through temporary hatch plate penetrations. The new installation will prevent failure of the penetration due to tugging on cord or cable. The SE determined that there were no unreviewed safety questions.

DCP 93-230

REMOVE CONTROL ROOM DOOR KNOB HARDWARE
(SE #93-SE-MOD-064)

DESCRIPTION

The door knob was removed at the entrance to the Unit 1 Control Room from the Turbine Building due to an identified safety problem. The door knob was very close to the air pressure seal and several individuals injured their hand when attempting to use the door knob. The door knob was not necessary because the lockset is electronically opened when a keycard is properly inserted. The door knob was replaced with a cover plate.

SUMMARY OF SAFETY ANALYSIS

The safety evaluation concluded that an unreviewed safety question did not exist. Issues considered include Control Room habitability and availability, fire integrity, and accessibility to the Control Room.

DCP 93-232

MODIFY TUBING SUPPORTS FOR 2-SI-FT-2945
(SE #93-SE-MOD-060)

DESCRIPTION

As a result of repetitive tubing failure at vent valve 2-SI-377 and the Low Head Safety Injection (LHSI) system pressure spike phenomenon, a complete system review was performed to determine if any other vent or instrument connections off of the LHSI system was vulnerable to failure from axial pipe movement. The review identified that tubing associated with flow transmitter 2-SI-FT-2945 could be enhanced by providing additional flexibility since original load cases did not include axial pipe movements associated with pump start. The modification removed three supports from the tubing for 2-SI-FT-2945 to provide additional flexibility. All piping allowable stresses were satisfied.

SUMMARY OF SAFETY ANALYSIS

The safety evaluation concluded that an unreviewed safety question did not exist since all pipe stresses were maintained within allowable limits.

DCP 93-007

MODIFICATIONS TO SUPPORT SG REMOVAL ON U2
(SE #93-SE-MOD-069)

DESCRIPTION

Various modifications were performed in the Unit 2 containment to facilitate future Steam Generator replacement. These changes included:

1. Relocation of electrical interferences.
2. Installation of anchor bolts for the runway beams and auxiliary crane support tower.
3. Installation of removable floor in equipment hatch barrel.
4. Modification of equipment hatch platform.
5. Laser templating of SG nozzles and reactor coolant piping.

The modifications do not impact the operation of any equipment in the plant.

SUMMARY OF SAFETY ANALYSIS

The safety evaluation considered the design basis of the containment and ensured that all modifications met required bases. The safety evaluation concluded that an unreviewed safety question does not exist.

SAFETY EVALUATION NUMBER 93-SE-MOD-065

DESCRIPTION

DCP 93-160-3 was written to replace 01-BR-57 & 59.

SAFETY EVALUATION SUMMARY

Existing check valves 01-BR-57 & 59 were replaced due to increasing problems. The old type check valves will be replaced with a new Circle Seal model. In addition the DCP will add new isolation valves in the line to pressure switch 01-BR-PS-600/601. The SE determined that there were no unreviewed safety questions.

SAFETY EVALUATION NUMBER 93-SE-MOD-066

DESCRIPTION

DCP 93-231 was written to address MCR ventilation dampers.

SAFETY EVALUATION SUMMARY

The MCR dampers are doors on the HVAC ducts which normally hang open. There have been a couple of incidents where the dampers have been close without direction. The DCP was written to modify the hatches such that they require positive personnel action in order to close them.

SAFETY EVALUATION NUMBER 93-SE-MOD-067

DESCRIPTION

DCP 93-008 was written to address Generic Letter 89-10 "MOV operation under specific circumstances.

SAFETY EVALUATION SUMMARY

Engineering wrote the DCP to address MOV's which are required to operate under certain specific circumstances be assured of proper operation. This includes the proper setting of thermal overloads. The new setting will optimize MOV availability and motor protection of the MOV's.

SAFETY EVALUATION NUMBER 93-SE-MOD-068

DESCRIPTION

DCP 93-013-3 Modifications for tornado loading of containment hatch platforms.

SAFETY EVALUATION SUMMARY

The DCP was written to add structural reinforcement to the safety related containment equipment hatch platforms, concrete repairs to containment hatch missile shields and provide a new connection detail between the roof and wall section of the missile shield. The SE determined that there were nc unreviewed safety questions.

SAFETY EVALUATION NUMBER 93-SE-MOD-070

DESCRIPTION

DCP 93-015-3 was written to ensure Rack Room cabinets do not create a problem during a seismic event.

SAFETY EVALUATION SUMMARY

Engineering determined that adjacent cabinets located in the instrument rack room need to be hooked together to ensure that during a seismic event the cabinets did not respond out of phase and impact one another such as to cause relay malfunctions. The SE determined that there were no unreviewed safety questions.

SAFETY EVALUATION NUMBER 93-SE-MOD-071

DESCRIPTION

DCP 93-182-2 Modify power to Aux. monitoring panel.

SAFETY EVALUATION SUMMARY

The power supply to Aux. monitoring panel 2-EI-CB-97A is beginning to show signs of a weakened condition. The battery charger is no longer needed for the monitoring cabinet because the batteries were removed by EWR 84-441A. It is impractical to maintain the current configuration. The SE determined that there were no unreviewed safety questions.

SAFETY EVALUATION NUMBER 93-SE-MOD-072

DESCRIPTION

DCP 92-360 was written to reduce the number of closure bolts on the fuel transfer tube blind flange.

SAFETY EVALUATION SUMMARY

This DCP was written to reduce the number of closure bolts on the fuel transfer tube blind flange from 20 to 4 bolts. In addition, a new blind flange will be machined for Quad rings will be substituted for the existing flange which utilizes zero rings. The purpose for the Mod was to reduce radiation exposure due to, the removal and installation of the flange during outages.

SAFETY EVALUATION NUMBER 93-SE-MOD-075

DESCRIPTION

DCP's 93-246 & 93-205 were written to replace tubing Tees off the pressurizer.

SAFETY EVALUATION SUMMARY

Tubing Tees off the PZR level transmitters, 01-RC-LT-1459,1460 &1461 and 02-RC-LT-2459,2460, &2461, are to be removed and replaced with bent tubing with unions or elbows. Bent tubing is the preferred method, however, depending on the room available elbows may be used instead. The Swagelock caps are not in accordance with plant specifications which require welded connections. The SE determined that there were no unreviewed safety questions.

SAFETY EVALUATION NUMBER 93-SE-MOD-079

DESCRIPTION

Changing the design temperature and pressure rating of the piping associated with the Steam Driven Aux. Feed water pump.

SAFETY EVALUATION SUMMARY

Engineering performed an evaluation of the turbine driven Aux. FW pump piping to change the design temperature to 100 degrees F and design pressure to 1480 psig. The relief valves lift point on both units will set at 1480 psig. This is being done to allow the RV's on the discharge of the pumps to be set at a higher setpoint. per original piping code requirements, RV set pressure is to be the same as the system piping design pressure. Currently, the RV's simmer when the pumps are operated on their full flow recic. lines. Failure of the RV renders the pump inoperable. The SE determined that there were no unreviewed safety questions.

SAFETY EVALUATION NUMBER 93-SE-MOD-080

DESCRIPTION

Station Blackout Diesel Generator Building Installation

SAFETY EVALUATION SUMMARY

DCP 92-010-3 was written by Engineering to address the installation of the SBO building. The DCP will install all underground utilities (fire protection, station air, fuel oil system) and electrical duct bank. The issue of effecting the flood berm was brought up and addressed by the DCP. No LCO's will be entered during the construction phase of the DCP. The SE determined that there were no unreviewed safety questions.

SAFETY EVALUATION NUMBER 93-SE-MOD-081

DESCRIPTION

DCP 93-160 was written to replace a check valve and install a new isolation for ease of maintenance of installed instrumentation.

SAFETY EVALUATION SUMMARY

Existing check valves (01-BR-57 & 59) are being replaced as a result of increased problems noted with valves. New isolation valves are being installed on the sensing lines to the gas stripper compressor leak detection pressure switch 01-BR-PS-600/601. In addition, new isolation valves will be installed in series with the check valves. The SE determined that there were no unreviewed safety questions.

SAFETY EVALUATION NUMBER 93-SE-MOD-082

DESCRIPTION

In order to reduce the time required to restore the RHR system in the event of a APP-R fire, 5KV breakers will be modified to supply RHR pumps.

SAFETY EVALUATION SUMMARY

DCP 93-276 was written to premodify a spare 5KV breaker on each unit in order to supply power to a unit's RHR pump if its normal power supply feeder breaker was lost due to an APP-R fire. These new breakers would be identified in the emergency procedures thus allowing for fast restoration of a RHR pump. SE determined that there were no unreviewed safety questions.

SAFETY EVALUATION NUMBER 93-SE-MOD-083

DESCRIPTION

U2's casing cooling recirculation pumps and chillers operate differently than U1's casing cooling pumps and chiller. Fix U1's system so it operates like U2's system because U2's system is more efficient.

SAFETY EVALUATION SUMMARY

A DCP was written to change the control circuitry of U1's casing cooling recirc. pumps and chillers. The design change will modify the control circuitry such that the pumps will run continuously via a local on/off switch. The new configuration will cause the chillers to cycle on temperature. In addition, the range on locally installed temperature indicators will be change to 0-75 degrees F. from 0- 150 degrees F. The SE determined that there were no unreviewed safety questions.

SAFETY EVALUATION NUMBER 93-SE-MOD-084

DESCRIPTION

Valves are currently installed horizontally. Manufacturer recommends the valves be installed vertically as they will not operate dependably in other positions.

SAFETY EVALUATION SUMMARY

Several component cooling water relief valves are to be reoriented so that they are installed vertically. A FC to DCP 93-102 was written to have maintenance install 02-CC-RV-213 in the correct orientation per the manufacturer's recommendation. The SE determined that there were no unreviewed safety questions.

DCP 92-014-2

STEAM GENERATOR SUPPORT HEATER REMOVAL
NORTH ANNA UNIT 2

DESCRIPTION

Due to the deletion of requirements for minimum and maximum temperature limits for the steam generator support beams, the need for the tent-like enclosures and resistance space heaters, which were required to achieve the elevated temperatures has been eliminated (Ref. Amendment No. 58. dated June 7, 1984 to Operating License). Therefore, the Unit 2 Steam Generator Support Heaters and associated equipment are no longer required and have been abandoned.

To gain access to the area under the steam generator in preparation for the eventual replacement of the Unit 2 Steam Generators, this package has disassembled and removed the steam generator support insulation tent material, support heaters, heater supports and cages and associated conduit and cables from Unit 2 containment. In addition, power, control and annunciator cables and an annunciator window have been spared outside the Unit 2 containment.

SUMMARY OF SAFETY ANALYSIS

This DCP was non-safety related, did not create an unreviewed safety question as defined by 10 CFR 50.59 and did not require a Technical Specification change.

A. The removal of abandoned steam generator support heaters and their associated in containment components will not adversely affect any other system in the plant. This system has previously been electrically disconnected and serves no function in the operation of the plant.

DCP 92-007

EDG FUEL OIL TRANSFER PUMP FLOW INSTRUMENTATION ADDITION
NORTH ANNA UNITS 1&2

DESCRIPTION

ASME Section XI, Subsection IWP, established the requirements for In-Service Testing (IST) of ASME Section III Class 1, 2, and 3 centrifugal and positive displacement pumps. Among these requirements was the measurement of flow and pressure with an accuracy of ± 2% of full scale. The design change involved installation of permanent flow measurement devices for the Fuel Oil Transfer Pumps (FOTPs) associated with the Emergency Diesel Generators (EDGs).

This activity provide installation of a bypass line containing isolation valves and a variable area flowmeter (vertical rotameter) in the discharge piping of each FOTP Since the bypass lines would be isolated except during pump testing, the normal operation and performance of the pumps was not affected. Installation of the flowmeters ensured that the ASME Section XI requirements for pump testing were met.

SUMMARY OF SAFETY ANALYSIS

Installation of the flowmeters did not constitute an unreviewed safety question as defined in 10CFR50.59 since it did not:

A) Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in UFSAR.

Since the modification only affected the measurement of FOTP flow to the EDG Day Tanks, it did not increase the frequency or consequences of accidents considered. The design change involved the addition of piping and components which would be isolated at all times other than during performance of Periodic Tests on the EDGs and FOTPs.

B) Create a possibility for an accident or malfunction of a different type than any evaluated previously in the UFSAR.

The change enhanced the quality of data taken during performance of existing periodic tests. The possibility of an accident of a different type than previously evaluated was not introduced by this work.

C) Reduce the margin of safety as defined in the basis of any Technical Specification.

This design change did not change the basis of any Technical Specifications. Implementation of the modification enhanced the margin of safety of the FOTPs by providing a more accurate indication of flow and thus overall condition.

DCP 91-002

Turbine Gland Steam System Upgrade
North Anna / Units 1 & 2

DESCRIPTION

The turbine gland steam systems at North Anna Power Station Units 1 and 2 have experienced extensive erosion/corrosion damage and had a considerable history of piping and component failures. This caused significant operational problems, and frequent maintenance activities.

A 1990 Westinghouse study on North Anna's turbine gland steam system indicated that the original gland steam supply flow associated with the high pressure turbine glands and high pressure spillover flow were understated by fifty percent.

Based on the results of the stress analysis performed for this design change, it was concluded that the discharge lines from the three gland steam supply lines relief valves were not adequately supported.

This design change replaced the turbine gland steam supply piping and valves from the gland steam supply header stop and bypass motor operated valves to each turbine gland connection and the excess steam spillover lines to the condenser. This design change changed the piping material from the carbon steel to Type 304 stainless steel, increased pipe sizes in specific locations, improved condensate drainage from the system, provided accessible Y-type stainless steel strainers, replaced existing carbon steel valves with new stainless steel valves, replaced undersized pressure control and motor operated valves with properly sized new valves, and replaced and upgraded existing system instrumentation and associated tubing. Also, the design change modified or added supports on the relief valve discharge lines to correct previously identified pipe support inadequacies. Minor relocation of interfering low pressure CO_2 fire protection piping in Unit 2 and minor rerouting of leak-off piping from Unit 1 main steam throttle valve 1-MS-TV-1C were also implemented under this design change.

SUMMARY OF SAFETY ANALYSIS

An unreviewed safety question was not created because:

The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because: The operation of safety-related equipment under normal or accident conditions was not affected by this design change. Also, the operation of the gland steam system as defined in the UFSAR was not changed or impacted by this design change. The modifications made were essentially a one-for-one replacement of components with enhanced design and superior material construction to preserve and prolong the current gland steam supply system.

The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because: This design change enhanced the turbine gland system's reliability by upgrade to more durable piping material and improved valve designs. The modifications were designed consistent with the original system's design basis.

The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because: The turbine gland steam system has no operational, design, or surveillance requirements in the Technical Specifications. No revision of the Technical Specifications was necessary as a result of this design change.

DCP 89-044

CONDENSER LAGGING REPLACEMENT
NORTH ANNA UNIT 1

DESCRIPTION

The feedwater heaters 1-FW-E-5A, 1-FW-E-5B, 1-FW-E-6A, 1-FW-E-6B and extraction steam piping located inside the main condenser were shrouded with stainless steel sheet metal lagging. The function of the lagging was to keep the heater and extraction line surfaces inside the condenser dry in order to avoid relatively large heat loss due to evaporative cooling. The heaters and extraction lines were originally lagged with 16 gauge stainless steel and insulation clips (stainless steel channel) were welded to the sheet metal lagging. The lagging was mounted on the clips to produce an annular space of one inch radial thickness.

The condenser inspection program detected reoccurring cracks and areas of significant lagging deterioration due to steam impingement and vibration induced fatigue. Also defects and gouges in the extraction steam piping and heater shells were discovered as a result of vibration between the sharp stainless steel standoffs welded to the lagging and the carbon steel extraction steam piping and heater shells.

The design change incorporated the use of 12 gauge stainless steel sheet metal to replace the 16 gauge in order to provide added rigidity for the lagging. Structural tube steel standoffs were welded to the heaters and extraction steam piping to provide lagging support and the required annular space. The package provide a more rigid lagging design which eliminated the vibration induced failures which were observed on the original lagging, piping and heaters.

SUMMARY OF SAFETY ANALYSIS

The replacement of the deteriorated lagging within the main condenser did not constitute an unreviewed safety question as defined in 10CFR50.59 since it did not:

A) Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in UFSAR.

The lagging replacement was essentially a one-for-one replacement with an enhanced design and did not affect any operations or ability of equipment important to safety to perform their safety functions.

B) Create a possibility for an accident or malfunction of a different type than any evaluated previously in the UFSAR.

New improved lagging design was developed in accordance with the latest requirements of HEI Standards for Steam Surface Condensers. The function of the new lagging did not change.

C) Reduce the margin of safety as defined in the basis of any Technical Specification.

This design change did not change the basis of any Technical Specifications. During implementation of the design change, the requirements of Section 3/4.7.1 were not violated. The section addresses the conditions required in the Main Steam System and Auxiliary Feed System for protection of the steam generator and to assure the capability to remove residual heat from the core during loss of station power. The lagging replacement was performed during unit refueling outage.

DCP-89-40-1
RTD BYPASS LINE ELIMINATION
NORTH ANNA POWER STATION

DESCRIPTION

The method of measuring the narrow range hot and cold leg reactor coolant temperatures used the resistance temperature detector (RTD) bypass system. This system was designed to address temperature streaming in the hot legs and to allow replacement of the direct immersion RTDs without draindown of the reactor coolant system (RCS). For increased accuracy in measuring the hot leg temperatures, mixing scoops were located in each hot leg at three locations of a cross section, 120 degrees apart. Each scoop had five orifices which sample the hot leg flow. The flow from the scoops is piped to a manifold where a direct immersion RTD measured the hot leg loop temperature upstream of the steam generator. The cold leg temperature was measured in a similar manner with piping to a separate bypass manifold, except that no scoops were used and only one location was utilized. Each hot leg and cold leg manifold contained three RTDs, two for use in the reactor protection system (one active and one spare), and one for use in the reactor control system.

The reactor coolant system RTD bypass lines were a significant contributor to personnel exposure due to their low flow velocity and the configuration of the piping, valves and manifolds which collect particulate corrosion (crud traps). The bypass lines were arranged at and around the level of the reactor coolant hot and cold legs, causing considerable exposure to personnel working in these areas (e.g., on reactor coolant pumps and steam generators, etc). Several plant unscheduled outages had been required for maintenance on components of the resistance temperature detector bypass lines. In addition, primary leakage through valve stem packing, flanges and manifolds had required additional maintenance during scheduled outages.

This modification consisted of the removal of the existing narrow range reactor coolant temperature measurement system including associated bypass manifold piping, valves, and manifolds and replacing it with a new narrow range temperature measurement system. The new system consists of thermowell mounted dual element, fast response, RTDs installed directly into the cold legs and modified scoops of the hot leg of the reactor coolant piping.

The RTDs are protection grade, and are wired to existing reactor protection instrument racks for ΔT and T_{avg} protection functions. The ΔT and T_{avg} signals are also wired through circuit isolators to the reactor control system for reactor control functions.

SUMMARY OF SAFETY ANALYSIS

This Design Change replaced the RCS narrow rang RTD bypass system with a thermowell mounted RTD temperature measurement system installed directly into the hot cold legs of the reactor coolant system. This modification was performed to increase plant availability and reliability by the removal of valves that have been the source of reactor coolant leakage inside the containment and also to reduce the man-rem exposures in keeping with corporate ALARA objectives.

The possibility of an accident or malfunction of a different type than any evaluated previously in the FSAR was not created. The changes were performed in a manner consistent with applicable standards, preserve the existing design bases, and did not adversely impact the qualification of any plant systems.

DCP-89-41-2
RTD BYPASS LINE ELIMINATION
NORTH ANNA POWER STATION

DESCRIPTION

The method of measuring the narrow range hot and cold leg reactor coolant temperatures used the resistance temperature detector (RTD) bypass system. This system was designed to address temperature streaming in the hot legs and to allow replacement of the direct immersion RTDs without draindown of the reactor coolant system (RCS). For increased accuracy in measuring the hot leg temperatures, mixing scoops were located in each hot leg at three locations of a cross section, 120 degrees apart. Each scoop had five orifices which sample the hot leg flow. The flow from the scoops is piped to a manifold where a direct immersion RTD measured the hot leg loop temperature upstream of the steam generator. The cold leg temperature was measured in a similar manner with piping to a separate bypass manifold, except that no scoops were used and only one location was utilized. Each hot leg and cold leg manifold contained three RTDs, two for use in the reactor protection system (one active and one spare), and one for use in the reactor control system.

The reactor coolant system RTD bypass lines were a significant contributor to personnel exposure due to their low flow velocity and the configuration of the piping, valves and manifolds which collect particulate corrosion (crud traps). The bypass lines were arranged at and around the level of the reactor coolant hot and cold legs, causing considerable exposure to personnel working in these areas (e.g., on reactor coolant pumps and steam generators, etc). Several plant unscheduled outages had been required for maintenance on components of the resistance temperature detector bypass lines. In addition, primary leakage through valve stem packing, flanges and manifolds had required additional maintenance during scheduled outages.

This modification consisted of the removal of the existing narrow range reactor coolant temperature measurement system including associated bypass manifold piping, valves, and manifolds and replacing it with a new narrow range temperature measurement system. The new system consists of thermowell mounted dual element, fast response, RTDs installed directly into the cold legs and modified scoops of the hot leg of the reactor coolant piping.

The RTDs are protection grade, and are wired to existing reactor protection instrument racks for ΔT and T_{avg} protection functions. The ΔT and T_{avg} signals are also wired through circuit isolators to the reactor control system for reactor control functions.

SUMMARY OF SAFETY ANALYSIS

This Design Change replaced the RCS narrow rang RTD bypass system with a thermoweil mounted RTD temperature measurement system installed directly into the hot cold legs of the reactor coolant system. This modification was performed to increase plant availability and reliability by the removal of valves that have been the source of reactor coolant leakage inside the containment and also to reduce the man-rem exposures in keeping with corporate ALARA objectives.

The possibility of an accident or malfunction of a different type than any evaluated previously in the FSAR was not created. The changes were performed in a manner consistent with applicable standards, preserve the existing design bases, and did not adversely impact the qualification of any plant systems.

DCP 89-04B

SERVICE AND INSTRUMENT AIR SYSTEMS UPGRADE
NORTH ANNA UNITS 1 & 2

DESCRIPTION

System failures caused by Instrument Air (IA) failures have occurred in the Nuclear Power Industry at a rate that indicated greater attention to Instrument Air systems was warranted.

The IA system provides compressed air primarily to air-operated control valves, such as the turbine control valves, secondary drains normal and high level divert valves, and feedwater and auxiliary feedwater control valves. Backup air flow capability is provided from the Service Air (SA) Subsystem during normal plant operation. The originally installed IA compressors were oil-free Ingersoll-Rand reciprocating compressors. The problem with the IA compressors was a combination of compressor type, environment , and available cooling water. The compressor was a reciprocating machine which inherently incurred higher maintenance than other designs due to the number of moving parts. This was aggravated by a rather warm operating environment resulting from surrounding equipment and the Heating Ventilation and Air conditioning (HVAC) design for the area. Compounding the problem, the compressors were cooled with Service Water. This cooling media caused corrosion of the coolant passage in the compressor. Corrosion buildup restricted the flow and deteriorated the heat transfer rate. The Service Water piping was replaced with stainless steel to alleviate plugging in the pipes.

The IA dryers (1/2-IA-D-1), are used to remove moisture in the air discharge by the instrument air receivers. The dryer(s) discharge is distributed to the instrument air loads as needed. The original IA dryers were Hankison refrigerant type dryers which did not meet the air quality requirements since they did not limit the particle size within the air steam to no greater than 3 microns and provide a pressure dewpoint 18 degrees fahrenheit below minimum design temperatures. The design indoor temperature is 50 degrees fahrenheit. With the ever increasing emphasis being placed on IA Systems, the Station committed to upgrade the performance of the Instrument Air Dryers to meet the quality requirements.

The SA Subsystem of the Compressed Air System provides compressed air to operate air-powered tools and equipment during normal operation and refueling.

It also acts as a backup for the IA Subsystem. In the original design, air was to be compressed by two 100 percent capacity compressors and stored in receiver tanks for use at tool and equipment connections located throughout the station. Pressure control valves between the receiver tanks were to provide a means to supply air from SA to IA.

The original specified SA compressors were oil-free Ingersoll-Rand reciprocating compressors. These machines were proved less reliable than required and were not operating because of damage due to insufficient cooling. The problems with the original SA compressors were the same as explained for the original IA compressors. The problems described for the IA System led to many compensatory actions in order to maintain suitable IA quality to the various components. These actions included the temporary installation of electric motor driven and diesel driven oil-free compressors along with temporary installation of a desiccant-type air dryer. Permanent improvements were required to address the problems noted for the IA system and ensure adequate IA quality for the remaining life of the plant.

To eliminate the problems noted above and also increase the reliability of the Compressed Air System, corrective actions were taken. The existing Sullair compressors 1/2-IA-C-5 were replaced with oil-free, air-cooled, rotary screw compressors and designated as SA compressors 1/2-SA-C-1. The existing service air compressors 1-SA-C-1 and 2-SA-C-1 were removed. The new air compressors doubled the capacity of the existing service air compressors and met the design basis capacity requirements of the original service and Instrument Air System.

The desiccant air dryer was relocated from the turbine building to a downstream flowpath of the receiver tank. In addition, a bypass line and associated bypass valve around the dryer, 1-IA-D-7, was added. The bypass valve included a solenoid operated actuator designed to open the bypass valve on low instrument air header pressure.

The instrument air compressors were replaced with more reliable oil-free, rotary screw, water-cooled compressors. The IA compressors remain powered from the emergency bus dedicated for emergency standby service for loss of station power events. The service water pressure boundary was maintained by heat exchangers which were commercial grade dedicated for safety related service.

The IA dryers were replaced with heatless desiccant dryers and associated pre and after filters. The replacement air dryers were sized for each to provide airflow capability for both units. The new dryers maintain a pressure dewpoint significantly lower than the original dryers.

DCP 89-04B-3 replaced 1-IA-C-1 which was removed under DCP 89-04A-3. After this was completed, demolition and replacement of 2-IA-C-1 was performed.

Demolition of dryer 2-IA-D-6 and compressor 2-IA-C-5 was completed and 2-SA-C-1 was then installed.

Installation of rebuilt dryer 2-IA-D-7 was completed and modifications were performed on associated piping and existing receiver tank 2-IA-TK-6.

SUMMARY OF SAFETY ANALYSIS

This Design Change did not constitute an unreviewed safety question as defined in 10CFR50.59, since it did not:

1. Increase the probability of occurrence of an accident or malfunction to safety as previously evaluated in the UFSAR. The reliability of the Compressed Air System was increased since the Service Air and Compressed Air compressor were replaced with more reliable air compressors. In addition, Instrument Air will be available during a loss of power event since the compressors would continue to receive power from the Emergency Power System.

2. Create a possibility for an accident or malfunction of a different type than evaluated in the UFSAR. This modification replaced the original two service air compressors with two new rotary screw air compressors capable of delivering the same air requirement as the original Service and Instrument Air compressors. In addition, the IA compressors were replaced with more reliable compressors and the IA dryers were capable of meeting air quality requirements sized for two unit operation.

3. Reduce the margin of safety as defined in the basis of any Technical Specification. The new service air compressors were not required for accident conditions and operation of the new compressors did not alter the function of any safety related system. The new IA compressors remain powered from the emergency bus.

1993 50.59 SAFETY EVALUATIONS REPORTABLE TO THE NRC

PRIOR TO USE (PTU) PROCEDURE REVISIONS

93-SE-PTU-001
93-SE-PTU-002
93-SE-PTU-003
93-SE-PTU-004
93-SE-PTU-005
93-SE-PTU-006
93-SE-PTU-007
93-SE-PTU-008
93-SE-PTU-009

SAFETY EVALUATION NUMBER 93-SE-PTU-001

DESCRIPTION

A B&W ultrasonic fuel inspection rig is to lifted out of the spent fuel pit while flushing. The High High radiation signal on the fuel building (FB) bridge trolley will have to be defeated.

SAFETY EVALUATION SUMMARY

A B&W Ultrasonic Fuel Testing rig is to be lifted out of the Spent Fuel pit. It was desired to placed the FB automatic interlock switch to defeat in the event the rig was contaminated and caused a High High radiation signal on the FB bridge and trolley crane radiation monitor. While the rig was being flushed, the FB automatic interlock switch was desired to remain in defeat while the rig remained suspended over the Spent Fuel pit. The SE determined that there was no unreviewed safety question.

SAFETY EVALUATION NUMBER 93-SE-PTU-002

DESCRIPTION

Control placing power range detector N42 in trip, reversing input cables A and B, performing voltage and current measurements, and restoring the detector back to normal.

SAFETY EVALUATION SUMMARY

Power range detector N42 was placed in trip, testing performed, and the detector restored to normal. This was performed to verify the condition of the instrument following calibration and testing on 3/21/93.

SAFETY EVALUATION NUMBER 93-SE-PTU-003

DESCRIPTION

Revise 1-OP-8.6, VCT Operation, to minimize oxygen addition to the Waste Gas Decay Tanks.

SAFETY EVALUATION SUMMARY

Control the connection of a stainless steel flex hose jumper between the VCT and process vent to allow the VCT to be purged directly to the process vent.

SAFETY EVALUATION NUMBER 93-SE-PTU-004

DESCRIPTION

Revise 1-MOP-50.02 to allow using PG to leak check CC pump seal package.

SAFETY EVALUATION SUMMARY

Using PG for a CC pump seal package leak check allows Ops the option of draining the water to a floor drain following the leak check. Jumper is only used during a maintenance activity.

SAFETY EVALUATION NUMBER 93-SE-PTU-005

DESCRIPTION

Provide justification for increasing alarm setpoints of certain ambient temperature alarms.

SAFETY EVALUATION SUMMARY

An Engineering review of the the nameplate ambient temperature ratings of all equipment in the areas of concern was performed. The most conservative temperature was chosen as the maximum allowable in that area.

SAFETY EVALUATION NUMBER 93-SE-PTU-006

DESCRIPTION

To allow for continued operation of the HP laundry during a Unit 2 Aux Steam tagout.

SAFETY EVALUATION SUMMARY

Revise 2-MOP-35.90 to add sections to the Aux Steam Master Tagout procedure to align steam the the HP laundry during a Unit 2 Aux Steam tagout.

SAFETY EVALUATION NUMBER 93-SE-PTU-007

DESCRIPTION

Revise 2-PT-210.19 to verify that the SI accumulator discharge valves are free to open and exhibit full stroke operation.

SAFETY EVALUATION SUMMARY

With fuel removed from the vessel and the vessel head off, a controlled dump of the SI accumulators into the RCS verifies that the discharge check valves exhibit full stroke.

SAFETY EVALUATION NUMBER 93-SE-PTU-008

DESCRIPTION

Revise 2-PT-210.19 to verify that the SI accumulator discharge valves are free to open and exhibit full stroke operation.

SAFETY EVALUATION SUMMARY

With fuel removed from the vessel and the vessel head off, a controlled dump of the SI accumulators into the RCS verifies that the discharge check valves exhibit full stroke.

SAFETY EVALUATION NUMBER 93-SE-PTU-009

DESCRIPTION

Revise 2-OP-8.6, ' CT Operation, to minimize oxygen addition to the Waste Gas Decay Tanks.

SAFETY EVALUATION SUMMARY

Control the connection of a stainless steel flex hose jumper between the VCT and process vent to allow the VCT to be purged directly to the process vent.

1993 50.59 SAFETY EVALUATIONS REPORTABLE TO THE NRC

SPECIAL TESTS

93-SE-ST-001
93-SE-ST-002
93-SE-ST-003
93-SE-ST-004

1993 SAFETY EVALUATION SUMMARY

SAFETY EVALUATION NUMBER 93-SE-ST-001

DESCRIPTION

The special test 1-ST-103 covers two independent phases of testing to facilitate replacement of the in-plant 450 MHz radio system with an 850 MHz system. The first phase specifically tests Unit 1 and Unit common areas for suspect Radio Frequency Interference (RFI) on instrument loops not located within the existing radio exclusion areas. The second phase of testing verifies that the radio reception and coverage will be acceptable in all required areas of the station within the existing posted RF restricted areas.

SAFETY EVALUATION SUMMARY

The Special Test 1-ST-103 is required to verify that the new 850 MHz radio trunking system will (1) provide improved coverage throughout the station and (2) will not cause adverse effects to the operation of safety related plant equipment control circuitry due to RFI.

1993 SAFETY EVALUATION SUMMARY

SAFETY EVALUATION NUMBER 93-SE-ST-002

DESCRIPTION

The special test 1-ST-103 covers two independent phases of testing to facilitate replacement of the in-plant 450 MHz radio system with an 850 MHz system. The first phase specifically tests Unit 1 and Unit common areas for suspect Radio Frequency Interference (RFI) on instrument loops not located within the existing radio exclusion areas. The second phase of testing verifies that the radio reception and coverage will be acceptable in all required areas of the station within the existing posted RF restricted areas.

SAFETY EVALUATION SUMMARY

The Special Test 1-ST-103 was issued to verify RFI impact (if any) by operating the new 850 MHz, 3 watt portable radios near plant instrumentation located in non-RFI restricted areas. This safety evaluation will support the PAR that will permit the testing of a 900 MHz, .521 milliwatt cordless phone in the Main Control Room and verify its RFI impact.

SAFETY EVALUATION NUMBER 93-SE-ST-003

DESCRIPTION

Special Test 1-ST-102 Feed Flow and Steam Generator Moisture Carryover Measurement Using Chemtrac Chemical Tracer Method
Purpose of the test was to determine the amount of moisture carryover from the "A", "B" and "C" Steam Generators to the Main Steam header and to validate the current steam flow instrument scaling.

SAFETY EVALUATION SUMMARY

The enriched Lithium-6 isotope (non-radioactive) injected into the feedwater train via normal system drain connections was validated by CME N-93-082 to have no negative affect on SG chemistry. Only one protection circuit cabinet was opened at a time to collect flow and pressure data. Test connections in the protection circuit cabinets were made at indication test points only and therefore isolated from any potential feedback to protective circuitry. All other data collection was via the GETARS and P-250 computer. The test required stable 100% unit power conditions, any ensuing transient required the suspension of the test until stable power conditions could be restored.

SAFETY EVALUATION NUMBER 93-SE-ST-004

DESCRIPTION

Special Test 2-ST-95 Feed Flow and Steam Generator Moisture Carryover Measurement Using Chemtrac Chemical Tracer Method
Purpose of the test was to determine the amount of moisture carryover from the "A", "B" and "C" Steam Generators to the Main Steam header.

SAFETY EVALUATION SUMMARY

The enriched Lithium-6 isotope (non-radioactive) injected into the feedwater train via normal system drain connections was validated by CME N-93-082 to have no negative affect on SG chemistry. Only one protection circuit cabinet was opened at a time to collect flow and pressure data. Test connections in the protection circuit cabinets were made at indication test points only and therefore isolated from any potential feedback to protective circuitry. All other data collection was via the GETARS and P-250 computer. The test required stable 100% unit power conditions, any ensuing transient required the suspension of the test until stable power conditions could be restored.

1993 50.59 SAFETY EVALUATIONS REPORTABLE TO THE NRC

OTHERS

91-SE-OT-039
92-SE-OT-066 Rev 4
92-SE-OT-086 Rev 1
92-SE-OT-101
93-SE-OT-001
93-SE-OT-002
93-SE-OT-003
93-SE-OT-004
93-SE-OT-005
93-SE-OT-006
93-SE-OT-007
93-SE-OT-008
93-SE-OT-009
93-SE-OT-010
93-SE-OT-011
93-SE-OT-012
93-SE-OT-013
93-SE-OT-013 Rev 1
93-SE-OT-014
93-SE-OT-015
93-SE-OT-016
93-SE-OT-017
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93-SE-OT-020
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93-SE-OT-036
93-SE-OT-036 Rev 1
93-SE-OT-037
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93-SE-OT-065
93-SE-OT-066
93-SE-OT-066 Rev 3
93-SE-OT-067
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93-SE-OT-093
93-SE-OT-094
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93-SE-OT-098

SAFETY EVALUATION NUMBER 91-SE-OT-039

DESCRIPTION

Increase the acceptance range for High Head Safety injection flow balance testing such that test failure due to instrument inaccuracies or lack of repeatable valve positioning is less likely and specifies a value for simulated RCP seal injection flow to be used.

SAFETY EVALUATION SUMMARY

This was a Tech Spec change that reduced the two lowest SI branch flows from greater than or equal to 384 gpm to greater than or equal to 359 gpm. Total SI flow was increased from less than or equal to 650 gpm to less than or equal to 660 gpm. In addition, a value of greater than or equal to 48.3 gpm is to be used for simulated RCP seal injection flow.

SAFETY EVALUATION NUMBER 92-SE-OT-066 Rev.4

DESCRIPTION

Repair/Replacement of 24" diameter Service water (SW) piping in U1 QS building

SAFETY EVALUATION SUMMARY

A DCP was written to repair/replace 24" SW piping in the QS building due to deteriorated condition from pitting corrosion. The DCP required isolation and draining of the affected SW loop. In addition, a 4" pipe cross connect was made between the Bearing Cooling (BC) and SW system to supply the U1 control room chiller when the SW system was not available.

SAFETY EVALUATION NUMBER 92-SE-OT-086, Rev. 1

DESCRIPTION

A Data-Logger will be connected to devices in the Main Generator Voltage Regulator Logic Drawer to help determine what intermittent problems exist in the voltage regulator.

SAFETY EVALUATION SUMMARY

The Data-Logger is a nonintrusive piece of test equipment which is designed to be connected to components or systems during normal operation such that continuous monitoring is possible. As such, it is designed to have a very high input impedance such that it will only monitor signals and not interact with the equipment being monitored. The Data-Logger is connected to the metering portion of the voltage regulator circuitry, not any portion of the control circuitry. This provides for a reasonable degree of safe operation of the voltage regulator. Should a malfunction in the Data-Logger cause a catastrophic failure of the voltage regulator, the ensuing unit trip is bounded by existing accident analyses.

SAFETY EVALUATION NUMBER 92-SE-OT-101

DESCRIPTION
0-ECM-0204-01 Installation of Temporary Residual Heat Removal Motor Feeder Cables

SAFETY EVALUATION SUMMARY
This procedure was written based upon the total loss of RHR capability due to a failure of power to the pumps due to an Appendix "R" fire. The procedure provides direction to route new cables to a new power supply. All of this operation is outside the bounds of both the UFSAR and Tech Specs, however, these actions will only be taken to mitigate the consequences of a severe fire that causes a total loss of RHR. Plant design is based on the assumption that certain accidents may occur and that sufficient redundant equipment is available to mitigate the damage. An Appendix "R" fire could cause a loss of equipment and all related redundancy. This places the station outside of all normal design bases and abnormal actions will be required. A total loss of RHR could cause an extremely severe accident and it would not be an unreviewed safety issue to return RHR to service in the most expeditious manner.

SAFETY EVALUATION NUMBER 93-SE-OT-001

DESCRIPTION

A 3 inch temporary spool piece will be installed between the main steam header and the main steam drain to the condenser to allow a temporary procedure to be implemented that will provide more time efficient cooldown of the steam generators.

SAFETY EVALUATION SUMMARY

The spool piece can be installed without creating an unreviewed safety question because:

- The spoolpiece will not be installed until after the plant is in Mode 5.
- The spool piece is located inside the main steam valve house and will not interfere with the ability to remove decay heat though the RHR system.
- The spool piece will remain in place during fuel movement in Mode 6. The auxiliary feedwater turbine steam line isolation and bypass valves upstream of the spool piece connection will be closed for each steam generator to satisfy the containment isolation requirements.
- If the spool piece failed during fuel movement and caused damage to the main steam piping, isolation valves, or bypass valves, core alterations can be suspended until containment integrity can be confirmed.
- The spool piece will be supported by beam clamps and rod hangers, placing no additional loads on the permanent main steam piping.
- The spool piece consists of materials of equivalent or greater pressure-retaining capacity than that of the main steam piping.
- Appropriate measures will be taken to prevent debris from entering the main steam system and temporary protection will be installed over the open valve bodies.

SAFETY EVALUATION NUMBER 93-SE-OT-002

DESCRIPTION

Added Evaporator Bottoms Tank low level and high/low temperature annunciator jumpers to defeat and restore alarms.

SAFETY EVALUATION SUMMARY

This modification disables unnecessary alarms for the Evaporator Bottoms tank when it is not being used to store boron, thus the tank's heaters are de-energized and do not need to be covered by minimum level, and the tank has been flushed with primary grade water. The ability to maintain boron concentration in the RCS by CVCS is unaffected.

SAFETY EVALUATION NUMBER 93-SE-OT-003

DESCRIPTION

Change the acceptance criteria for the integrated leak rate test from 0.75 La to 1.0 La for the "as-found" leak rate only thus reducing the potential for additional testing which is unnecessary although required by Appendix J.

SAFETY EVALUATION SUMMARY

The "as-left" leak rate criteria will remain at 0.75 La to maintain the operational and analysis margin. Containment leakage during a design basis accident is assumed to be 1.0 La for the first hour of the accident. After the first hour the containment is assumed to be maintained subatmospheric and no additional leakage is assumed.

SAFETY EVALUATION NUMBER 93-SE-OT-004

DESCRIPTION

0-TOP-49.01 0-TOP-49.02

The Temporary Operating Procedures (TOP) allow for the A and the B Service Water headers to be removed from service to install blank for header repair.

SAFETY EVALUATION SUMMARY

Repairs are required on the Service Water headers. In order to repair the Service Water headers on Unit 1 during the Unit 1 outage, blanks must be installed to isolate the Unit 1 portion under repair from the Unit 2 portion of the headers which will be in operation.

SAFETY EVALUATION NUMBER 93-SE-OT-005

DESCRIPTION

0-TOP-49.01

Attach a stem blocking device to 1-SW-TV-101A and 101B.

SAFETY EVALUATION SUMMARY

Service water system modifications and repairs are being made that require removing 1-SW-MOV-110A, 110B, 114A, and 114B. 1-SW-TV-101A and 101B are the isolation valves between Service Water and the Recirc Air Heat Exchangers. These trip valves fail to the closed condition, but Maintenance Engineering has determined that a blocking device should be installed to assure the valves remain closed. It is desired to maintain Chilled Water available to the Containment Recirc Air Heat Exchangers during SGRP work; therefore, 1-CC-TV-115A and 115B will not be closed to provide backup isolation capability.

SAFETY EVALUATION NUMBER 93-SE-OT-006

DESCRIPTION

A UFSAR change to delete settlement monitoring of survey points that are rock founded and/or have shown no significant movement.

SAFETY EVALUATION SUMMARY

Associated with Tech Spec amendments 167 and 147, most of the original Tech Spec settlement monitoring points did not need to be monitored. This change deleted the unnecessary points from the UFSAR.

SAFETY EVALUATION NUMBER 93-SE-OT-007

DESCRIPTION

Revision to PT's and ICP's for the power range NI's to ensure that the time limit or Tech Spec 3.3.1.1 is not violated during the performance of the testing.

SAFETY EVALUATION SUMMARY

ICP's and channel functional PT's for the power range NI's were revised to change the way in which the channels are place in test and the way the bistable functions are tested and verified. The procedure change was an enhancement that allowed all functions of the NI channel to be tested in accordance with the Tech Specs.

SAFETY EVALUATION NUMBER 93-SE-OT-008

DESCRIPTION

This SE provide an evaluation of Fire Barriers with pipe insulation passing through them.

SAFETY EVALUATION SUMMARY

Station Deviation Report N93-115 &116 identified inoperable fire barriers due to insulation on the pipe which went through the fire barrier. An Engineering evaluation determined that the barrier noted in the DR's and other similar barriers are operable and do not require a fire watch.

SAFETY EVALUATION NUMBER 93-SE-OT-009

DESCRIPTION

The Safeguards building sump pump motors were incorrectly added to the Q-List as SR.

SAFETY EVALUATION SUMMARY

Procurement Engineering evaluated the reclassification of the Safeguards Building sump pump motors from SR to non-safety related. The SE determined that there were no safety concerns or unreviewed safety questions.

SAFETY EVALUATION NUMBER 93-SE-OT-010

DESCRIPTION

T.S. 3.1.3.1 change to clearly define control and shutdown rod banks which are trippable and misaligned by no more than 18 steps as operable for up to 72 hours.

SAFETY EVALUATION SUMMARY

This was a tech spec change and was approved by the NRC as such. The rods remain fully trippable and will therefore perform their intended safety function. All current core design limits will continue to be met.

SAFETY EVALUATION NUMBER 93-SE-OT-11

DESCRIPTION

T.S. clarification then EDG is operable in modes 5 and 6 while paralelled to the emergency bus but not operable when in this configuration during modes 1-4.

SAFETY EVALUATION SUMMARY

Operator action can be taken credit for in modes 5 or 6 to restore the EDG (if it trips) or to manually load the necessary components on the emergency bus. This is consistent with UFSAR and NUREG assumptions.

SAFETY EVALUATION NUMBER 93-SE-OT-012

DESCRIPTION

Cancellation of FCAs which govern fires in the steam driven aux feed pump house, turbine building and aux service water pump house.

SAFETY EVALUATION SUMMARY

Each function of the FCA procedures to be deleted have been evaluated to verify that the actions under the FCA are redundant to actions provided under other existing procedures.

SAFETY EVALUATION NUMBER 93-SE-OT-013

DESCRIPTION

Replacement of 2-SD-LS-249A. De-energize the solenoids for 2-SV-TV-200A&B. This fails open the normally open SOVs, sustaining MSR drainage to the "A" 1st point FW heater. In addition, the hi level alarms for the 1A, 3A and 4A FW heaters will be disabled.

SAFETY EVALUATION SUMMARY

Only the high level divert auto function is defeated; the probability of a secondary transient during this maintenance is extremely small. The hi-hi level protection remains unaffected.

SAFETY EVALUATION NUMBER 93-SE-OT-013, Rev. 1

DESCRIPTION

In order to replace 2-SD-LS-249A, 2-SV-TV-200A will be failed open and the high level trip functions of several SD and SV valves, as well as part of the water induction system interlock of the 1A, 3A, and 4A FW heaters will be disabled.

SAFETY EVALUATION SUMMARY

2-SV-TV-200A is only a minor contributor to the inventory of the 1A FW heater and will be failed open for only a few hours. Although the trip functions of the reheater drain receiver NLC valves, the high level divert valves, and the 3A to 4A cascade valve will be defeated, the valves will still modulate on demand, sustaining level stability in the FW heaters and the Drain Receivers. 2-SV-TV-200C may still be closed by pushbutton if required. The ES NRVs and TVs will still move to their required positions on 1) a Turbine Trip from the venting of the ES air header by the air pilot valve located on the governor valve emergency trip fluid header, or 2) a FW heater hi-hi level signal (originating from an alternate level switch). In addition, the FW heater hi-hi level functions remain operable to secure ES via the ES MOVs to protect the turbine from water induction.

SAFETY EVALUATION NUMBER 93-SE-OT-014

DESCRIPTION

This Tech Spec change allows, as an alternative to tube plugging, the option to repair degraded sections of steam generator tubes by laser welded tube sleeving. The change also permits recovery of tubes previously plugged for corrective or preventive measures.

SAFETY EVALUATION SUMMARY

Testing has proven the use of laser welded tube sleeving to provide a leaktight bond between the sleeve and the tube during all plant conditions. Non-destructive examination of the sleeve length and non-sleeved tube section still can be performed. Any combination of sleeving and plugging up to the level that the minimum measured reactor coolant flow is maintained per the Tech Specs will be bounded by the accident analyses supporting the analyzed flow rate.

SAFETY EVALUATION NUMBER 93-SE-OT-015

DESCRIPTION

This UFSAR change clarifies protective coating requirements used in the containment building in sections 3.8.2.7.6 and 6.2.2.2.

SAFETY EVALUATION SUMMARY

This change clarifies the requirements for protective coatings used inside the containment building, but no changes are being made to existing requirements.

SAFETY EVALUATION NUMBER 93-SE-OT-016

DESCRIPTION

This SE evaluates core reload and operation of NAPS Unit 1 for Cycle 10.

SAFETY EVALUATION SUMMARY

Core reload has been evaluated against existing accident analyses and found to be within the design and licensing bases.

SAFETY EVALUATION NUMBER 93-SE-OT-017

DESCRIPTION

This SE evaluates removal of admin locks from auxiliary feedwater pump individual full flow recirculation lines.

SAFETY EVALUATION SUMMARY

Removing the admin locks on full flow recirc lines will not reposition any valves nor will the operation of the system be affected. The lock will be retained on the common return line to the ECST.

SAFETY EVALUATION NUMBER 93-SE-OT-018

DESCRIPTION

This SE evaluates operation of the CC system as a contaminated system.

SAFETY EVALUATION SUMMARY

Analysis of the design and supporting features of the CC system indicate that operation as a contaminated system, although not originally intended, was provided for and presents no radiological or operability concerns.

SAFETY EVALUATION NUMBER 93-SE-OT-019

DESCRIPTION

This UFSAR revision adds and clarifies the operating signals to liquid waste discharge valve 1-LW-PCV-115.

SAFETY EVALUATION SUMMARY

This system is non-safety related, including all related valve signals. Operation of this valve has no impact on the consequences or mitigaion of any previously analyzed accident.

SAFETY EVALUATION NUMBER 93-SE-OT-020

DESCRIPTION

The following valves will be stroked against a delta-P with VOTES data recorded to satisfy the requirements of GL 89-10:
1-CH-MOV-1115B, C, D, E
1-CH-MOV-1267A, B, C
1-CH-MOV-1269A, B, C
1-CH-MOV-1270A, B

SAFETY EVALUATION SUMMARY

The CH system and MOVs will be operated within design limits and IAW approved procedures and the UFSAR. The delta-P against which the CH MOVs will be tested is within the MOV design calculated delta-P conditions. The test will be properly monitored by Operations and System Engineering. Operations always has the option to abandon the test should unanticipated problems arise. Tech Specs will be complied with during the test.

SAFETY EVALUATION NUMBER 93-SE-OT-021

DESCRIPTION

The following valves will be stroked against a delta-P with VOTES data recorded to satisfy the requirements of GL 89-10:

1-CH-MOV-1275A, B, C
1-CH-MOV-1286A, B, C
1-CH-MOV-1287A, B, C
1-CH-MOV-1289A, B
1-CH-MOV-1373
1-CH-MOV-1370

SAFETY EVALUATION SUMMARY

The CH system and MOVs will be operated within design limits and IAW approved procedures and the UFSAR. The delta-P against which the CH MOVs will be tested is within the MOV design calculated delta-P conditions. The test will be properly monitored by Operations and System Engineering. Operations always has the option to abandon the test should unanticipated problems arise. Tech Specs will be complied with during the test.

SAFETY EVALUATION NUMBER 93-SE-OT-022

DESCRIPTION

This SE evaluates the classification of the Decay Heat Release Valve Hand Control Operator based on a review of the pertinent design and licensing documents to identify any adverse consequences of downgrading the operators to NSQ.

SAFETY EVALUATION SUMMARY

The operators are presently SR in the Q-List. They are being changed to NSQ due to their function. A review of the licensing documentation does not reveal any safety related functions for these operators.

SAFETY EVALUATION NUMBER 93-SE-OT-023

DESCRIPTION

The RHR RVs have a lift setpoint of 467 psig. The current springs are suitable for a relief setting of 411 to 450 psig.

SAFETY EVALUATION SUMMARY

Based on engineering evaluation, it was determined that use of the existing springs on an interim basis would be acceptable and that they would be replaced during the next outage.

SAFETY EVALUATION NUMBER 93-SE-OT-024

DESCRIPTION

This SE evaluates removal of terminal strip safety covers that obstruct banana jack adapters in the process racks and NI cabinets.

SAFETY EVALUATION SUMMARY

A review of the industry standards for terminal strips/blocks (UL Standard 1059 and ANSI/NEMA Standard ICS4) indicates that these covers are for identification purposes and are not subject to operability criteria. Engineering was contacted at Marathon Special Products Division. It was stated that these covers are typically installed for personnel safety concerns.

SAFETY EVALUATION NUMBER 93-SE-OT-025

DESCRIPTION

Tri-axial cables from the Unit 1 C and D Incore Detectors will be disconnected and used for installation of temporary remote cameras in the Unit 1 containment.

SAFETY EVALUATION SUMMARY

The Incore Movable Detector System is not required during outage conditions. Remote camera monitoring is used for various activities in containment during the outage, including monitoring of the local RCS standpipe. The electrical jumper will consist of using two tri-axial cables from the incore detector system with the new remote camera system.

SAFETY EVALUATION NUMBER 93-SE-OT-026

DESCRIPTION

This SE evaluates stroking the following valves against a delta-P and recording VOTES data to satisfy the requirements of GL 89-10:
1-SI-MOV-1862A, B
1-SI-MOV-1863A, B
1-SI-MOV-1885A, B, C, D
1-SI-MOV-1890A, B, C, D
1-SI-MOV-1864A, B

SAFETY EVALUATION SUMMARY

Uncontrolled Boron dilution is not credible in plant mode 6. Sampling of RWST and SI header liquid for boron concentration assures sufficient boron concentration is maintained for shutdown margin. No PG water path exists which could dilute the header boron concentration. VCT overpressurization is not likely since the VCT will be isolated from the test boundary. The path via the Seal Water Heat Exchanger to the VCT is isolated by shutting 1-CH-213 and 1-CH-214 and backseating 1-CH-215. The RH and RP systems will be operable during this test. A boration path is maintained at all times. No penetration to any RCS boundary will occur to conduct this test.

SAFETY EVALUATION NUMBER 93-SE-OT-027

DESCRIPTION

This SE supports qualification and release of an updated RETRAN model for the performance of non-LOCA transient analysis for licensing and operational support.

SAFETY EVALUATION SUMMARY

The model consists of several thousand individual input parameters which are descriptive of the plant. The basis for each parameter is documented in a detailed model input notebook, including reference sources, auxiliary calculations performed to develop the input parameters, and cautions and limitations associated with the application of each aspect of the model. The notebook is subjected to the same peer review and quality assurance requirements as a NAF calculation, thus producing a comprehensive data base and reference volume which will be available to all users of the model. The model has been qualified via a process which is equivalent, in scope and rigor, to the qualification of the original models. Calculation have been performed with the updated model and the results compared with the earlier RETRAN model for the following UFSAR transients:
- Uncontrolled RCCA Bank Withdrawal at Power
- Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition
- Complete Loss of Reactor Coolant Flow
- Loss of External Electrical Load
- Loss of Normal Feedwater
- Locked Reactor Coolant Pump Rotor

The bases for the selection of the transients used in the qualification were:
- transients determined to have been limiting from prior analyses
- transients in each of the major categories of initiating events such as reactivity change, variations in primary coolant rate and changes in primary to secondary heat transfer rate
- both symmectric and asymmetric transients with respect to the response of the RCS loops
- the full range of initial conditions, i. e., HZP, HFP with and without deterministic instrument error effects

The results showed that the updated model's behavior is generally consistent with that of the earlier model, and differences in the results can be explained in terms of modeling refinements for the updated model. In particular, the updated model uses a more physically accurate representation of the Doppler reactivity feedback. In the earlier representations, a Doppler power coefficient as a function of core heat flux was often used. In the updated model, Doppler feedback is represented as a function fuel temperature, which is consistent with the physical basis for the phenomenon. Use of the temperature coefficient makes the overall dynamic response of the system to transients more sensitive to the core thermal modeling assumptions.

SAFETY EVALUATION NUMBER 93-SE-OT-028

DESCRIPTION

Flange leakage has been identified on the line immediately below the #3 seal leakoff line on the Unit 2 "C" reactor coolant pump. A Furmanite collar and injection repair will be performed on the flange.

SAFETY EVALUATION SUMMARY

This repair will seal the flange leak. The flange bolts will be replaced, ensuring the integrity of the connection. The Furmanite repair is a commonly used maintenance evolution. Further, this maintenance is not a pressure boundary repair; it is at a gasketed flange connection, so that there are no Generic Letter 90-05 concerns (Non-Code Repairs to Piping).

SAFETY EVALUATION NUMBER 93-SE-OT-029

DESCRIPTION

Internal engineering reviews have identified some of the contacts between the Manual SI and its input to the Reactor Trip Breakers as having not been tested as per Tech Spec 4.3.1.1.1 Table 4.3-1, Item 19. This missed surveillance is governed by TS 4.0.3, necessitating a make-up surveillance within 24 hours. Unit 2 is presently operating at 100% power; because an at-power test would require an entry into TS 3.0.3, disable the Manual SI and the subsequent Reactor Trip on Manual SI, a waiver of compliance will be sought until the test can be performed during the next refueling outage.

SAFETY EVALUATION SUMMARY

No credit is taken in the safety analyses for the RPS input from the manual SI signal. Testing performed via existing surveillances verified that between the two manual SI switches all reactor trip and bypass breakers are verified tripped. Emergency Operating Procedures direct the operator to actuate both manual SI switches. Further, both the Emergency Operating Procedures and operator training require the operators to verify the reactor trip breakers are open prior to manually initiating SI. Finally, sufficient redundancy exists via the Reactor Trip on Automatic SI signals to compensate for the untested contacts.

SAFETY EVALUATION NUMBER 93-SE-OT-030

DESCRIPTION

This Tech Spec change separates the subsystems of the containment recirculation spray system into two clearly defined trains. This separation allows clear and concise actions to be stated for the loss of one or more of the subsystems or a train of containment recirculation spray.

SAFETY EVALUATION SUMMARY

The containment recirculation spray system works in conjunction with the quench spray system to depressurize the containment building following a design basis accident. This change will not affect the containment recirculation spray systems capability of performing its design function for the following reasons:
1) The recirculation spray system is comprised of four 50% capacity subsystems. A loss of one subsystem (inside or outside spray pump) still provides 150% capacity. A loss of one train (one inside pump, one outside pump, and its respective casing cooling pu p) still provides 100% capacity for the accident analysis.
2) Changing the Tech Spec 3.6.2.2 to establish the separation of the recirculation spray system into two trains agrees with the UFSAR and the System Design Basis Document terminology for accident analysis.
3) Tying the casing cooling subsystem into its associated outside recirculation spray pump is consistent with the purpose of the casing cooling pump which is to supply an adequate amount of cold water to the suction of the outside recirculation spray pump for NPSH concerns.

SAFETY EVALUATION NUMBER 93-SE-OT-031

DESCRIPTION

Blocks will be installed to maintain 1-RH-HCV-1758 and 1-RH-FCV-1605 throttled, and 1-CH-TV-1204A and 1-CH-HCV-1142 open during performance of the Type A test since instrument air will be isolated to containment.

SAFETY EVALUATION SUMMARY

During performance of the Type A test, Instrument Air to containment will be isolated. In order to maintain RHR sampling capability, letdown capability, and improve RHR temperature control, various valves will have blocks installed. While the blocks are installed, they will not be able to be manipulated from the control room. RHR temperature will be controlled via throttling CC to the RHR heat exchangers. 1-CH-TV-1204A is a containment isolation valve which closes on a Phase A signal. This function is not required while the unit is in mode 5. In addition the penetration can be isolated by the isolation valve outside of containment. Once the blocks are installed and valve position verified to be correct the chances for failure of the device is small. The blocks being installed will weigh approximately 1% of the valve/actuator in all applications except for 1-CH-TV-1204A. The blocking device for this valve weighs approximately 10-15% of the valve/actuator weight. This was analyzed and it was concluded since the valve has a support adjacent to it it is considered rigidly supported and impact of the block on seismic integrity is negligible. Since the other blocks add such a small amount of weight to the valve/actuator the effect on the seismic integrity of the lines they are in is also negligible. If the block on 1-RH-HCV-1758 were to fail the valve would go full open allowing more water to pass through the RHR heat exchanger. Tenperatures would be controlled by reducing CC flow to the RHR heat exchangers. If the block on 1-RH-FCV-1605 were to fail the valve would go closed. This would reduce the RHR flow but would not result in a total loss of flow. Alternate core cooling will also be available in the form of natural circulation, reflux boiling, and forced feed and bleed since the PORVs will be blocked open. Therefore the heat removal capability will not be significantly reduced. If the blocks on the remaining valves were to fail the valves would go closed causing a loss of letdown capability. This would not be detrimental to the RCS since overpressure protection is available via the blocked open PORVs and RHR RVs.

SAFETY EVALUATION NUMBER 93-SE-OT-032

DESCRIPTION

Internal engineering reviews have identified some of the contacts between the Manual SI and its input to the Reactor Trip Breakers as having not been tested as per Tech Spec 4.3.1.1.1 Table 4.3-1, Item 19. This missed surveillance is governed by TS 4.0.3, necessitating a make-up surveillance within 24 hours. Enforcement discretion was granted that allowed this surveillance to be deferred until the next refueling outage. This temporary change will incorporate the enforcement discretion into Tech Specs.

SAFETY EVALUATION SUMMARY

No credit is taken in the safety analyses for the RPS input from the manual SI signal. Testing performed via existing surveillances verified that between the two manual SI switches all reactor trip and bypass breakers are verified tripped. Emergency Operating Procedures direct the operator to actuate both manual SI switches. Further, both the Emergency Operating Procedures and operator training require the operators to verify the reactor trip breakers are open prior to manually initiating SI. Finally, sufficient redundancy exists via the Reactor Trip on Automatic SI signals to compensate for the untested contacts.

SAFETY EVALUATION NUMBER 93-SE-OT-033

DESCRIPTION

The proposed Tech Spec change affects Surveillance Requirements 4.4.5.2, 4.4.5.4, and 4.4.5.5 and their associated Tech Specs Bases. The proposed change provides, as an alternative to plugging, the option to repair degraded sections of steam generator tubes by sleeving. The proposed change adds an acceptance criteria definition for "Tube Repair" to Surveillance Requirement 4.4.5.4. The change also defines the sleeve plugging limit, measured as a percentage of degradation through the wall of the sleeve, which would require that the tube (sleeved tube assembly) be removed from service. The proposed license amendment package also includes several editorial type changes found on the affected Tech Spec pages, e. g., relocating commas and correcting typographical errors.

SAFETY EVALUATION SUMMARY

Tube repair by sleeving is accomplished by attachment of a smaller diameter length of thermally treated Alloy 690 tubing to the inside surface of a defective mill annealed Alloy 600 steam generator tube. The repair is applicable to both tube support plate intersections and within the tubesheet area. At the tube support plate intersections, the sleeve is first hydraulically expanded at the ends of the sleeve in order to bring the tube and sleeve to contact for optimization of the weld. An autogenous laser weld is produced within the hydraulically expanded regions. For repairs within the tubesheet, the sleeve is similarly hydraulically expanded at each end. The upper end of the sleeve extends past the top of the tubesheet, thereby spanning all areas where steam generator tubes have historically experienced degradation associated with the tubesheet. An autogenous laser weld is also produced within the upper expansion region of the tubesheet sleeve. At the lower end, the sleeve design prevents the bottom of the sleeve from extending completely into the tube end. A mechanical roll expansion is performed which supplies the necessary structural and leaktight integrity to the joint. As an option, a laser seal weld can be produced at an elevation coincident with the approximate midpoint of the tubesheet cladding. This weld functions only as a backup leakage barrier since the roll expansion has been shown to supply the necessary structural and leakage integrity to the joint. The installed sleeve will return the defective tube to a condition consistent with the design basis of the plant with regard to leakage and rupture considerations. The sleeve and sleeve attachment joints have been designed and analyzed according to Section III of the ASME Boiler and Pressure Vessel Code. Both structural and fatigue evaluations were performed. The installed sleeve free span (out of tubesheet) joints are initially inspected using an ultrasonic inspection technique to verify that the minimum weld fusion zone thickness which satisfies the structural and fatigue evaluation criteria is achieved. A baseline eddy current inspection is also performed prior to operation. The sleeves described in the Westinghouse Reports (WCAP-13088, Rev. 1) have been designed, analyzed, or tested to meet the service requirements of the Series 44 and 51 steam generators through the use of conservative and enveloping thermal boundary conditions and structural

Rev. 1, and WCAP-13619), the laser welded sleeves are concluded to meet applicable ASME Boiler and Pressure Vessel Code and regulatory requirements for North Anna Unit 2. Mechanical leakage testing of the sleeve and the sleeve joints is over and above the ASME Code and regulatory requirements. This testing is primarily concerned with leak resistance and joint strength. Even though the mechanical leak testing was performed at temperatures slightly less than anticipated hot leg temperatures (reference Section 4.0 of WCAP-13088, Rev. 1), the test pressure condition was far more severe than normal operating and in excess of postulated accident conditions for North Anna Unit 2. Testing under conditions more severe than any anticipated plant operating or accident condition have shown the laser welded joints satisfy all structural requirements and are leaktight during all anticipated operating and accident conditions. Testing has also shown that the lower joint of the tubesheet sleeve need not necessarily be welded in order to exhibit leaktight characteristics. A sample of the non-welded lower tubesheet test joints were subjected to steam line break loading prior to leak testing. Leak testing of these non-welded tubesheet sleeve lower joints has shown the sleeve to be leaktight during test at anticipated operating and accident conditions. The margin of safety with respect to maintenance of the integrity of the tube bundle is provided, in part, by the safety factors included in the ASME Code and is not reduced. Non-destructive examination of the sleeve length and non-sleeved tube section still can be performed. Therefore, the Tech Spec required tube inspections can still be implemented. The installation process of laser welded sleeves has been shown to provide a leaktight bond between the sleeve and the tube during all plant conditions, and as such would not contribute to the radiological consequences of a postulated steam line break event. Any combination of sleeving and plugging utilized at North Anna Unit 2 up to the level that the minumum measured reactor coolant flow rate is maintained per the Tech Spec requirements, will be bounded by the accident analyses supporting the analyzed flow rate.

SAFETY EVALUATION NUMBER 93-SE-OT-034

DESCRIPTION

Following maintenance on 1-SI-89, the cold leg check valve will be flow tested using a hydraulic jumper from the "C" accumulator to a LMC just upstream of the check valve.

SAFETY EVALUATION SUMMARY

The test will be performed when neither the ECCS subsystems nor the accumulators are required to be operable. An operator will be stationed at the LMC in order to isolate it in the event of a jumper failure. The dilution of the RCS is not a concern since RCS and accumulator boron concentrations will be obtained and adequate shutdown margin verified. The amount of air which may be injected into the RCS will be approximately 40 cubic feet. This air will be injected into the cold leg and will travel down the downcomer and be dispersed as it travels up the core by the lower internals and fuel assemblies. This will be adequate to prevent a slug of air from entering the suction of the RHR pumps which could result in a loss of RHR. If a loss of RHR were to occur, Alternate Core Cooling is available. Any voiding in the reactor head as a result of the air can be vented as required and will be detected via RVLIS.

SAFETY EVALUATION NUMBER 93-SE-OT-035

DESCRIPTION

Revise the UFSAR to: (A) clarify the surveillance requirements for fire detection control panels, (B) resolve inconsistencies with the require action between Unit 1 and 2 for hose stations, (C) correct a typographical error in the penetration fire barriers section, and (D) revision of the fire protection design basis to include Appendix R, Chapter 12.

SAFETY EVALUATION SUMMARY

The changes in the UFSAR do not increase the probability for a fire to occur. Consequences from a fire are not changed since these changes do not degrade the fire protection plan and ensure continued compliance with Appendix R and Appendix A to BTP 9.5.1. Consequences are still bounded by the Appendix R analysis. No accident of a different type is created since fires are still bounded by the original assumptions for Appendix R and Appendix A (single fire criteria). The margin of safety is not changed since fire protection will provide the level of protection required to ensure continued compliance with Appendix R and Appendix A to BTP 9.5.1.

SAFETY EVALUATION NUMBER 93-SE-OT-036

DESCRIPTION

Approval of 1(2)-MOP-31.5 for operation of Feedwater Control Valves using Manual Override.

SAFETY EVALUATION SUMMARY

1(2)-MOP-31.5 was written to allow manual override of feedwater control valves upon malfunction of a Main Feed Reg Valve. Override is used only because a MFRV is unstable; the MOP is not used for routine at-power maintenance. The alternative to performing maintenance with the valve in Override is to leave the MFRV in its degraded condition. Use of manual override is usually limited to a short period of time, typically less than 12 hours. The automatic isolation feature of the MFRVs may be defeated for MFRV maintenance by considering the following features:

1) The main FW pumps trip on a SI or SG hi-hi level.
2) The fast-acting MOVs auto-close on a SI or a P-14 signal (Train A).
3) Immediate actions in 1(2)-E-0 require verification of FW isolation following a SI, with explicit instructions to verify closure of the fast-acting MOVs and removal from service of all MFW pumps.
4) If a Reactor trip occurred without a SI, 1(2)-ES-0.1 directs the operator to verify that the MFRVs close at 554 F.
5) Only one valve at a time will be placed in override.
6) Procedure 1(2)-FR-H.3 can be entered if SG levels exceed 75% NR. The procedure verifies FW isolation.
7) Tech Specs require a limiting FW isolation time of 8.0 seconds. Compliance with the TS will be maintained via closure of the fast-acting MOVs and tripping of the MFW pumps, on a FW isolation signal, when a MFRV is in Override. These features are tested to ensure TS compliance every 18 months.
8) A MFRV may be placed in Override for a maximum of 72 hours, consistent with the TS action time of ESF components. This limit must be administratively imposed and maintained.

SAFETY EVALUATION NUMBER 93-SE-OT-036, Rev. 1

DESCRIPTION

Approval of 1(2)-MOP-31.5 for operation of Feedwater Control Valves using Manual Override.

SAFETY EVALUATION SUMMARY

1(2)-MOP-31.5 was written to allow manual override of feedwater control valves upon malfunction of a Main Feed Reg Valve. Probabalistic calculations documented in STA Calculation 93-01 have shown that although Override immobilizes the MFRV, the perturbation in the total failure probability of the MFRV is negligibly small. This assessment was based upon actual plant history and a review of the NSAC-125 criteria for evaluating 10CFR50.59 accident probabilities. Second, Override is used only because a MFRV is unstable; the MOP is not used for routine at-power maintenance. The alternative to performing maintenance with the valve in Override is to leave the MFRV in its degraded condition. It is plainly preferable to immobilize the erratic MFRV for the short repair period than to continue to operate with deteriorated SG level control. Consistent with the allowed outage time for ESF equipment, a valve may be placed in Override for a maximum of 72 hours per occurrence (consistent with the single train action limit of Standard Tech Specs). Further, only one valve at a time may be placed in Override. Finally, the following plant and procedural features provide redundant assurance of FW isolation:

1) The fast-acting MOVs auto-close on a SI or a P-14 signal (Train A).

2) The MFW pumps trip on a SI or P-14 signal.

3) The MFW pump discharge MOVs auto close on their respective FW pump trip; the discharge MOV of the standby pump may be manually closed.

4) Dedicated operators are stationed at both the control board and the MFRV to ensure that the valve will be closed when a FW isolation is required.

5) Immediate actions in 1(2)-E-0 require verification of FW isolation following a SI, with explicit instructions to verify closure of the fast-acting MOVs and removal from service of all MFW pumps.

6) If a Reactor trip occurred without a SI, 1(2)-ES-0.1 directs the operator to verify that the MFRVs close at 554 F.

7) Procedure 1(2)-FR-H.3 can be entered if SG levels exceed 75% NR. The procedure verifies FW isolation.

8) Tech Specs require a limiting FW isolation time of 8.0 seconds. Compliance with the TS will be maintained via closure of the fast-acting MOVs and tripping of the MFW pumps, on a FW isolation signal, when a MFRV is in Override. These features are tested to ensure TS compliance every 18 months.

The fast-acting MOVs are powered off of the SS 4kV buses. In addition, the SSPS input to the FW isolation on the 154s (254s) receives a signal from Train A only; the two motors on each FW pump receive a trip signal from different trains. Under the limiting MSLB scenario, the fast-acting MOVs will lose power and fail in the non-conservative postion without providing FW isolation. In this case, loss of the MFW pumps (via SI or loss of power) will provide the necessary FW isolation (along with

.SLB, it is possible that the fast-acting MOV on the Overridden feedline will not close, but feedflow will continue due to the possibility of one or more condensate pumps running on the still available SS busses. This postulated scenario would potentially result in mass addition to the faulted SG in excess of that assumed in the safety analysis. The probability of this scenario has been assessed in STA Calculation 93-01 and found to be negligible.

MSLB, it is possible that the fast-acting MOV on the Overridden feedline will not close, but feedflow will continue due to the possibility of one or more condensate pumps running on the still available SS busses. This postulated scenario would potentially result in mass addition to the faulted SG in excess of that assumed in the safety analysis. The probability of this scenario has been assessed in STA Calculation 93-01 and found to be negligible.

SAFETY EVALUATION NUMBER 93-SE-OT-037

DESCRIPTION

Document the Surveillance Requirement Position concerning the operability of Containment Hydrogen Recombiner purge blowers and the containment purge blowers 1/2-HC-F-1.

SAFETY EVALUATION SUMMARY

Westinghouse Standard Tech Specs state that a hydrogen purge cleanup system will be available if there are less than two hydrogen recombiners available. The standard design for this type of system consists of a purge blower, HEPA filters, charcoal adsorbers, and heaters to maintain moisture quality in the adsorbers. By Standard Tech Specs, this system must also be capable of being initiated from the control room. This type of system is not installed at North Anna since there are two hydrogen recombiners available. The UFSAR description of the containment purge blower is that it can be operated in parallel with the hydrogen recombiner system blowers when the containment is to be purged, ensuring that a failure of the recombiner system will not leave the contaiment without purge capability (UFSAR Section 6.2.5.2, Page 6.2-157). The backup containment purge blower is designed as a Non-Q system; that is, it can be assumed to be unavailable following a design basis earthquake. The backup purge blower is not part of the hydrogen recombiner system. It in no way interfaces with the hydrogen recombiner nor can it be lined up to supply the hydrogen recombiners. Though the containment purge blowers 1/2-HC-F-1 are a part of the containment atmospheric cleanup system described in the UFSAR, they are not a part of the hydrogen recombiner system and they have no bearing on the operability of the recombiners. The North Anna Tech Specs include specific Tech Specs for the hydrogen analyzers and another for the hydrogen recombiner. The hydrogen recombiner Tech Spec includes a requirement that each purge blower is operated at least once per six months for a minimum of 15 minutes. It is the position of North Anna that 1/2-HC-F-1 are not the purge blowers required by Specification 3/4.6.4.2 since they are redundant and parallel to the skid mounted hydrogen recombiner. Based upon the UFSAR description, the North Anna specific Tech Specs, and the Standard Westinghouse Tech Specs that require a separate purge blower system for plants with less than two recombiners, the Tech Spec statement "and that each purge blower operates for 15 minutes" implies that the blower associated with each hydrogen recombiner is the component that is required to be operable. Since Tech Spec 3.6.4.2 is specifically written for the recombiner system and the recombiner system contains a purge blower, it is reasonable to conclude that the hydrogen recombiner system surveillance requirement refers to the purge blower that is integral to the hydrogen recombiner.

SAFETY EVALUATION NUMBER 93-SE-OT-038

DESCRIPTION

This is a Tech Spec change to revise the main steam safety valve setpoint tolerance from 1% as found/1% as left to 3% as found/1% as left and modify the bases of TS 3/4.7.1.2 to address a reduction in the minimum AFW system delivered flow rate to 300 gpm/pump.

SAFETY EVALUATION SUMMARY

This change will serve to eliminate violations of the MSSV setpoint Tech Spec surveillance criteria by taking advantage of inherent margins in the safety analyses which are sensitive to the MSSV lift setpoint to justify a higher allowable setpoint tolerance. Because the increased MSSV lift setpoint tolerance could result in increased AFW system flow backpressure and, therefore, in a reduced AFW delivered flow rate, credit has been taken for flow margin in existing safety analyses to support a reduction of the assumed delivered AFW system flow rate to 300 gpm. The basis statement of TS 3/4.7.1.2 is being modified to reflect this reduction in the minimum AFW delivered flow rate, expand the discussion on pump surveillance tests and clarify the basis statement. An increased MSSV setpoint tolerance was explicitly modelled in the four UFSAR transients identified to be affected by the proposed setpoint tolerance increase. The Primary and Secondary side overpressurization results of the Loss of External Electrical Load and Locked RCP Rotor events remained well within their respective acceptance criteria. The Loss of Normal Feedwater and Main Feedline Break transient analyses assumed an AFW flow rate consistent with a 3% tolerance/3% accumulation MSSV opening characteristic. The key safety criteria for these two transients were demonstrated to be met. The calculations of OTdT setpoints were shown to be unaffected by the setpoint tolerance increase. Further, operational margin was shown to not be adversely impacted by the setpoint tolerance increase.

SAFETY EVALUATION NUMBER 93-SE-OT-039

DESCRIPTION

This SE evaluates 0-ECM-0102-04 as a replacement for EMP-C-BY-2.1 and changes the scope to only allow a maximum of two Main Station Battery cells to be equialize charged at one time while the battery is on float.

SAFETY EVALUATION SUMMARY

The number of individual cells that can be charged at once is limited to two. The charger will be located in the Battery Room which has little traffic. The leads are insulated and the connection points to the battery cells are separated by a minimum of one foot to prevent accidental shorting. The charger will be restrained to prevent motion in a seismic event. The battery room ventilation system is adequate to prevent an explosive accumulation of hydrogen during the individual cell charging. Should the ventilation system fail during individual charging, insufficient hydrogen will collect to cause an explosive concentration before corrective action can take place. An explosion or fire will be isolated within the battery room and will not affect other equipment.

SAFETY EVALUATION NUMBER 93-SE-OT-040

DESCRIPTION

The overexcitation relay setpoint for the emergency diesel generators is to be changed in the UFSAR from 135 amps to a range of 59.4 to 62.1 amps.

SAFETY EVALUATION SUMMARY

The UFSAR currently states that the EDG overexcitation relay is set at 135 amps, which is 250% of the full load current. This information is incorrect. The relay setpoint range should be 59.4 to 62.1 amps which corresponds to 110 to 115% of full load current according to calculation EE-00504. This is the current installed relay setting. This change is an editorial change only to correct an error in the UFSAR.

SAFETY EVALUATION NUMBER 93-SE-OT-041

DESCRIPTION

EWR 85-225 removed the heat tracing annunciator panel from the main control room. UFSAR sections 7.7.1.12.2 and 6.3.2.1.5 are changed by this EWR.

SAFETY EVALUATION SUMMARY

This EWR is classified as non-safety related. Removal of the heat tracing annunciator panel from the main control room does not impact the ability of the heat tracing to perform its function. Monitoring the status of the heat tracing is performed by LOG-6D (and LOG-6B and -6E) at the individual heat tracing annunciator cabinets. This meets the Tech Spec surveillance requirements.

SAFETY EVALUATION NUMBER 93-SE-OT-042

DESCRIPTION

In order to make the Operational Quality Assurance Program Topical Report consistent with VPAP-1801, the following changes will be made:

- Remove all references to "observations"
- clarify the definition of subjective evidence
- correct a typographical error
- delete the requirement for follow-up responses or the waiving of that requirement.

SAFETY EVALUATION SUMMARY

This revision to the QA Topical Report affects the auditing process only. There are no modifications being made to the plant. The design of the plant is not affected.

SAFETY EVALUATION NUMBER 93-SE-OT-43

DESCRIPTION

This SE evaluates allowing the backup computer train of ERFCS to be disabled for approximately 4 months to support reliability improvements.

SAFETY EVALUATION SUMMARY

During the period of this SE, if one of the computers fails, it will not affect the redundant computer since it is in Halt. Therefore, the back-upcomputer can be immediately started to restore the ERF computer system. This activity actually increases the redundancy of the ERF computer by not allowing a communications failure on the On-line machine to affect the machine placed in Halt. This computer configuration is temporary and the normal On-line back-up configuration will be restored. The ERFCS was designed as a passive monitoring device. The Emergency Response Teams use the computer system in assisting with plant safety assessment. In addition to providing data for safety assessment, it stores pre and post transient data which is used to determine if the plant systems responded as required to a specific event. The ERF/SPDS computer system is only a secondary assessment tool. The system is not discussed in Tech Specs, nor is it safety related.

SAFETY EVALUATION NUMBER 93-SE-OT-044

DESCRIPTION

Qualification/release of an improved PDQ model for the performance of reload core design neutronic analysis.

SAFETY EVALUATION SUMMARY

The two zone PDQ model represents an improvement over currently used models for core reload design. Enhancements include the use of more recent ENDF/B-V nuclear data, explicit modeling of multiple fission product chains as well as 3-D and 2-D configurations. The PDQ two zone models have been developed using state of the art methodology and qualified in the same manner as those previously approved for core design and core follow calculations. The conservatisms established for existing models are bounding for the two zone model predictions. These models are essentially equivalent replacements for those currently being used and their use will not modify the currently approved methodologies for performing reload core design and safety evaluations.

SAFETY EVALUATION NUMBER 93-SE-OT-045

DESCRIPTION

0-ECM-0101-02, "Installation or Removal of Jumper Cables Over Individual Cells of Main Station and Emergency Diesel Generator Stationary Batteries"

SAFETY EVALUATION SUMMARY

This Safety Evaluation is for an upgraded procedure which provides direction for installing jumpers on Station Batteries and EDG Batteries.
The procedure is useful for maintaining the operability of Station Batteries and EDG Batteries while performing maintenance on individual cells.

SAFETY EVALUATION NUMBER 93-SE-OT-046

DESCRIPTION

PT-14.6 provides guidance to pressurize the LHSI piping in order to determine the size of the gas bubble within the piping. Once the bubble size is known the header can be vented and the line repressurized to determine the effect of the venting process.

SAFETY EVALUATION SUMMARY

Operation of the vent valves will be in a normal fashion for liquid system vents. Operator action will ensure any vent open at the time of a Safety Injection Signal will be immediately closed. The amount of liquid vented will be insignificant and will have no operability impact on the RWST or LHSI system. The liquid charged into the LHSI system will be from the RWST, therefore there is no dilution concern. Pressurization of the LHSI piping will be to nominal pump recirc pressure. Relief valves in the lines will ensure overpressure protection. Tech Specs will be complied with. Therefore there is no safety concern.

SAFETY EVALUATION NUMBER 93-SE-OT-047

DESCRIPTION

Tech Spec change request #293 reduces from two to one the number of steam generators (S/G) required to be opened and inspected during the first refueling after S/G replacement. The change also removes extraneous information from the table (Table 4.4-1; Tech Spec 4.4.5.1), removes a note and renumbers the table notes.

SAFETY EVALUATION SUMMARY

The revision will only reduce the number of S/Gs opened for inspection during the first cycle. The number of tubes inspected will be the same (i.e. the minimum tube sample size in one S/G is twice that of two S/Gs). Each of the S/Gs is manufactured and operated in the same manner and the performance history of the Westinghouse S/Gs has shown that degradation of the tubes is not significant in early cycles of operation. Therefore the probability of occurrence of an accident is not increased. The revised inservice inspection program will continue to provide adequate detection of tube degradation, therefore there is no new type of failure which will result due to this change.

SAFETY EVALUATION NUMBER 93-SE-OT-048

DESCRIPTION

Large Break LOCA analysis was revised to assume up to 20% steam generator (S/G) tube plugging in any S/G. This was performed to support full power operation of North Anna units 1 and 2.

SAFETY EVALUATION SUMMARY

The analysis results show that the ECCS will meet the acceptance criteria as presented in 10CFR50.46. All analysis parameters were equivalent to, or conservative with respect to, those allowed by Tech Specs. All analysis parameters are expected to be conservative with respect to actual conditions for North Anna units 1 and 2. The analysis demonstrated that calculated results meet all design acceptance criteria as stated in the UFSAR. Therefore the margin of safety will not be reduced as a result of increasing S/G tube plugging up to 20%.

SAFETY EVALUATION NUMBER 93-SE-OT-048

DESCRIPTION

Large Break LOCA analysis was revised to assume up to 20% steam generator (S/G) tube plugging in any S/G. This was performed to support full power operation of North Anna units 1 and 2.

SAFETY EVALUATION SUMMARY

The analysis results show that the ECCS will meet the acceptance criteria as presented in 10CFR50.46. All analysis parameters were equivalent to, or conservative with respect to, those allowed by Tech Specs. All analysis parameters are expected to be conservative with respect to actual conditions for North Anna units 1 and 2. The analysis demonstrated that calculated results meet all design acceptance criteria as stated in the UFSAR. Therefore the margin of safety will not be reduced as a result of increasing S/G tube plugging up to 20%.

SAFETY EVALUATION NUMBER 93-SE-OT-049

DESCRIPTION

Revision to UFSAR section 5.5.8.4 to reflect new testing commitments for the pressurizer PORVs based on response to NRC Generic Letter 90-06.

SAFETY EVALUATION SUMMARY

Based on the response to Generic Letter 90-06, North Anna committed to testing the control air system valves for the PORVs. This required a change to the UFSAR since it currently states "no further test program is considered necessary".

The accidents evaluated in the UFSAR will not be affected by this change. The added testing will enhance the availability of the PORVs. Testing of the valves will not result in any new type of accident or failure.

SAFETY EVALUATION NUMBER 93-SE-OT-050

DESCRIPTION

Tech Spec change request #286 includes administrative, editorial, and technical changes in support of the revised 10CFR20 and reduces the reporting frequency of the radiological effluent release report from semi-annually to annually. In addition, Figure 5.5-1 is being revised to correctly identify the unrestricted area for gaseous effluents.

SAFETY EVALUATION SUMMARY

The Tech Spec change is in support of the revised 10CFR20 and Reduction of Regulatory Burden on Nuclear Licensees. This change does not affect the plant design or operation. As a result, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated nor does it increase the probability of an accident.

SAFETY EVALUATION NUMBER 93-SE-OT-051

DESCRIPTION

Tech Spec change request #288 which removes the frequencies for audits listed in Tech Specs. The change is based on the subject audit frequencies being governed by a summary statement in Virginia Power Operational Quality Assurance Program Topical Report, VEP 1-5A. The proposed change would also delete the Tech Spec requirement for SNSOC review of the PSP and EP.

SAFETY EVALUATION SUMMARY

The likelihood that an accident will occur is no affected by this Tech Spec change. Frequency of performance of program audits is not a precursor to or cause of any accident analyzed in the UFSAR. There are no consequences to equipment since the change does not have any impact on the design or operation if any plant equipment. There is no Bases section for Section 6 of Tech Specs. Since no equipment or safety limits are affected, the margin of safety is not affected.

SAFETY EVALUATION NUMBER 93-SE-OT-052

DESCRIPTION

Tech Spec change request #297 which removes the schedular requirement for Type A tests to be performed at a 40 +/- 10 month interval from TS 3.6.1.2 and adds that Type A testing will be performed in accordance with Appendix J to 10CFR50. Other editorial and administrative changes are also made.

SAFETY EVALUATION SUMMARY

Due to the 18 month fuel cycle for North Anna it is difficult to perform the required three tests within 10 years when limited by the added stipulation of 40 +/- 10 months in between tests. The proposed change will allow the flexibility to perform the three tests within the 10 year requirement as required by 10CFR50 Appendix J. The test type, method and acceptance criteria will not be changed. Therefore the bases of the testing is not affected and there is no detrimental impact on any accident analyses.

SAFETY EVALUATION NUMBER 93-SE-OT-053

DESCRIPTION

Tech Spec change request #294 will add a new Basis section to TS 6.5.2.7 to clarify the responsibilities of the MSRC regarding their review of 1) safety evaluations prepared pursuant tc 10CFR50.59 requirements and those prepared for other purposes and 2) SNSOC meeting minutes and reports.

SAFETY EVALUATION SUMMARY

The MSRCs review of safety evaluations required by 10CFR50.59 and overview of remaining safety evaluations ensures that reviews for unreviewed safety questions and consideration of nuclear safety issues are being properly addressed. The effectiveness of the safety evaluation program and the thoroughness of SNSOC meetings and reports will be assured through the MSRC overview function. This review only provides a check of the quality of the program and therefore has no negative impact on nuclear safety.

SAFETY EVALUATION NUMBER 93-SE-OT-054

DESCRIPTION

Tech Spec change request #296 (TS 6.8.2) deletes the requirement to periodically review most types of administrative and technical procedures. Procedures which have review frequencies based upon regulatory requirements, which are event driven or symptom based will continue to be reviewed as required by regulations.

SAFETY EVALUATION SUMMARY

Deletion of the review process of various procedures will not result in an increase in the consequences of an accident nor would it increase the probability of an accident. This change is in line with the regulatory reduction effort. The change is administrative in nature and has no impact on the operation or function of SR equipment.

SAFETY EVALUATION NUMBER 93-SE-OT-055

DESCRIPTION

Revision to the QA Topical Report which consists of clarifications of definitions, revision of organizational structures, administrative corrections, deletion of various reviews of procedures and incorporation of various changes to streamline the audit process.

SAFETY EVALUATION SUMMARY

The changes to the Topical Report are administrative in nature and have detrimental no impact on SR equipment or procedures. Neither operation of the plant nor its design are affected. The deletion of various procedure reviews will reduce the commitment to, but not the effectiveness of, the Operational QA Program.

SAFETY EVALUATION NUMBER 93-SE-OT-056

DESCRIPTION

1/2-AP-22.5, Loss of Emergency Condensate Storage Tank 1/2-CN-TK-1 (With Three Attachments) DR N-93-1078, Possible Overpressurization of Auxiliary Feedwater Pump Discharge Piping

This is an evaluation of the change to 1/2-AP-22.5, Loss of Emergency Condensate Storage Tank 1/2-CN-TK-1 (With Three Attachments) that requires use of the full-flow recirc line to control flow to the Steam Generators and limit discharge pressure to <1400 psig.

SAFETY EVALUATION SUMMARY

DR N-93-1078 identifies the possibility of overpressurizing the discharge piping of the Auxiliary Feedwater Pumps when the alternate water sources are supplied by Service Water or Fire Protection Water due to higher pressure experienced at the AFW Pump suctions. This additive pressure from the pump providing alternate suction produces a worst case discharge pressure of approximately 1550 psig when supplied by the Fire Protection System's Motor Driven Fire Pump.

SAFETY EVALUATION NUMBER 93-SE-OT-057

DESCRIPTION

Temporary Procedure (1-TOP-8.3) was developed to provide guidance to makeup to the VCT during maintenance on 01-CH-219 and 01-CH-FCV-1114A (normal makeup valves).

SAFETY EVALUATION SUMMARY

The flowpaths used for the makeup will consist of permanently installed piping and equipment. Indication of the boric acid solution and PG flows will be available to the OATC. The boration flowpath required by Tech Spec 3.1.2.2 will remain operable during the entire evolution. All makeups will be manually performed, therefore operations personnel will be cognizant of the evolution and any problems which may arise.

SAFETY EVALUATION NUMBER 93-SE-OT-058

DESCRIPTION

Revision of the UFSAR and the applicable procedures which require the EDG exhaust to be aligned through the muffler during EDG testing. The revision will delete this requirement.

SAFETY EVALUATION SUMMARY

The EDG muffler is bypassed while the EDG is in auto. The exhaust is aligned through the muffler during performance of PTs on the EDGs. There have been misposition events in which the muffler was not completely bypassed following a PT. Sound level surveys were performed to determine the difference between the decibel level when the EDG muffler is in service and when it is bypassed. The results did not show a significant difference. Therefore the muffler bypass valve will be locked open all the time. The EDG operation is enhanced since the potential for mispositioning of the muffler bypass has been eliminated.

SAFETY EVALUATION NUMBER 93-SE-OT-059

DESCRIPTION

Tech Spec change request #282B to allow repair of degraded sections of S/G tubing by sleeving versus plugging them. Tech Specs 4.4.5.2, 4.4.5.4, and 4.4.5.5.

SAFETY EVALUATION SUMMARY

The proposed change allows for repair of S/G tubes vs plugging them. The change also allows for recovery of tubes previously plugged for preventative or corrective measures. The proposed change adds an acceptance criteria for tube repair and defines the sleeve plugging limit.

Since the hydraulic impact upon the RCS of a sleeve is much less than a plug, use of sleeves minimizes the total number of effectively plugged tubes which will prolong the life of the S/G. By maintaining additional reactor coolant flow area through the S/Gs, sleeving can provide margin to the LOCA accident analyses assumptions.

SAFETY EVALUATION NUMBER 93-SE-OT-060

DESCRIPTION

0-OP-4.2, Rev. 3 UFSAR Chapter 9.1.1

The UFSAR, Chapter 9, states that "Unirradiated fuel assemblies are normally stored in a plastic wrapper to maintain cleanliness." The plastic wrapper has caused problems when loading new fuel into the storage tubes by catching on the tube and pulling grid straps off of the fuel assemblies. It is desired to remove the plastic wrapper prior to storing new fuel.

SAFETY EVALUATION SUMMARY

The use of a plastic wrapper to maintain cleanliness during new fuel storage is no longer a necessity. New fuel is not stored for long periods of time. Also, the new fuel storage area is maintained clean and the new fuel storage tubes have sealing lids which further promote a clean environment. Removal of the plastic wrapper will prevent unnecessary damage of the new fuel assemblies. This SE supports changing both the OP and the UFSAR.

SAFETY EVALUATION NUMBER 93-SE-OT-061

DESCRIPTION

Update of Appendix "R" Report due to completion of design change packages (DCP) and incorporation of engineering evaluations.

SAFETY EVALUATION SUMMARY

Engineering evaluations were performed in accordance with Generic Letter 86-10 in reference to use of Thermo-Lag 330-1 material as an Appendix "R" radiant energy shield and as a one hour rated fire barrier and for two new penetration seal configurations. The evaluations concluded that they provide adequate separation based on the hazards in the areas addressed. Other updates were as a result of modifications per DCPs. The changes were document updates only and are in compliance with 10CFR50 Appendix R.

SAFETY EVALUATION NUMBER 93-SE-OT-062

DESCRIPTION

Request for discretionary enforcement for Tech Spec Surveillance Requirement 4.5.2.h.1.a and 4.5.2.h.1.b.

SAFETY EVALUATION SUMMARY

Repairs and refurbishment of 1-CH-P-1C and its effects on the pump performance. Use of flow balance specifications proposed in Technical Specification Change Request #259 in lieu of existing specifications 4.5.2.h.1.a and 4.5.2.h.1.b. 1-CH-P-1C has been refurbished to repair damage to the pump's rotating element which potentially changes the pump head curve. Tech Spec surveillance requirements require that a a flow balance be performed following modifications to the ECCS system that alter the system flow characteristics. Enforcement discretion is needed to revise the acceptable band for HHSI flow balance, thus providing a high degree of certainty that the installed rotating element will meet the revised flow specifications when the pump is tested at the next refueling outage.

SAFETY EVALUATION NUMBER 93-SE-OT-063

DESCRIPTION

Performance of test boring in the unit 2 alleyway to determine the actual foundation conditions at the Service Water piping/Service Building interface relative to the documented foundation conditions. This is being performed to assess documented settlement values obtained by optical survey.

SAFETY EVALUATION SUMMARY

Extensive subsurface investigation has been performed to ensure that there is no SR equipment in the location of the boring. Performance of this evolution will have no impact on any SR equipment. Upon completion, the holes will be filled in and the asphalt patched.

SAFETY EVALUATION NUMBER 93-SE-OT-064

DESCRIPTION

Temporary shielding will be placed over select RCS piping during the 1993 unit 2 refueling outage. The shielding will reduce dose rates in the area where work is to be performed.

SAFETY EVALUATION SUMMARY

Calculations have been performed to show that the piping on which the blankets are placed will not fail during a seismic event. In addition all lead blankets will be wrapped around the pipe or hung off of Tub-Lok or nearby structural members in a manner such that they will not damage nearby SR equipment.

SAFETY EVALUATION NUMBER 93-SE-OT-065

DESCRIPTION

Tech Spec 5.3.1 and 6.9.1.7.e are being revised to allow for use of ZIRLO fuel cladding.

SAFETY EVALUATION SUMMARY

Upon irradiation, the ZIRLO fuel cladding exhibits improved corrosion resistance and dimensional stability over the current Zircaloy-4 cladding. The change in chemical composition of the cladding required evaluation of the potential impact on neutronic models and methods, primarily due to the presence of niobium. The net effect of these changes on reactivity and power distribution is small when compared to other modeling uncertainties.

The new cladding also effects some mechanical properties such as clad growth and clad creep rates which affects fuel temperatures and rod internal pressures. Chapter 15 accidents were re-evaluated and it was concluded that 10CFR50.46 acceptance criteria will continue to be met.

SAFETY EVALUATION NUMBER 93-SE-OT-066

DESCRIPTION

Delta-P testing of various LHSI MOVs (LHSI pump suction, discharge and recirc MOVs and HHSI suction from LHSI discharge MOVs) in response to NRC Generic Letter 89-10.

SAFETY EVALUATION SUMMARY

The MOVs will be stroked against design differential pressure conditions to verify that they will operate under design conditions. The test will be performed with the reactor head removed. All of the valves and system components will be operated within design conditions and Tech Specs will be complied with during the test. Sampling of the RWST/SI header prior to the test will ensure shutdown margin is maintained. The VCT will be isolated from the test boundary therefore the potential for overpressurization does not exist.

SAFETY EVALUATION NUMBER 93-SE-OT-66, Rev 3

DESCRIPTION

DCP 91-009-1, Repair and partial replacement of the 24" SW lines to the Unit 1 QS building.

SAFETY EVALUATION SUMMARY

A probabilistic Risk Assessment was performed for the periods which SW would not be available to the operating unit (the other was defueled) and showed the activity would add a negligible amount of increase to the frequency of core damage events. Relief was given to missle shileding requirements for the buried SW lines and electrical ducts. Further probability of core damage was reduced by providing an alternate supply for the cooling water for the control room chillers (bearing cooling).

SAFETY EVALUATION NUMBER 93-SE-OT-067

DESCRIPTION

Delta-P testing of various unit 2 Charging system MOVs (charging pump suction and RWST isolation MOVs) in response to NRC Generic Letter 89-10.

SAFETY EVALUATION SUMMARY

The MOVs will be stroked against design differential pressure conditions to verify they will operate under design conditions. The test will be performed with the unit in Mode 5 or 6 and RCS not solid or in a reduced inventory condition. All of the valves and system components will be operated within design conditions and Tech Specs will be complied with during the test. VCT pressure will not exceed 50 psig during the test. Its design pressure is 75 psig therefore there is no increase in potential for its rupture due to the test. Procedural and administrative blocking of the primary grade water flowpath is sufficient to ensure a dilution does not occur.

SAFETY EVALUATION NUMBER 93-SE-OT-068

DESCRIPTION

Delta-P testing of various unit 2 Charging system MOVs in response to NRC Generic Letter 89-10.

SAFETY EVALUATION SUMMARY

The MOVs will be stroked against design differential pressure conditions to verify they will operate under design conditions. The test will be performed with the unit in Mode 5 or 6 and RCS not solid or in a reduced inventory condition. All of the valves and system components will be operated within design conditions and Tech Specs will be complied with during the test. Procedural and administrative blocking of the primary grade water flowpath is sufficient to ensure a dilution does not occur.

SAFETY EVALUATION NUMBER 93-SE-OT-069

DESCRIPTION

NA-C-DCO-807 Switchyard procedure to support maintenance on Transformer #3 (34.5-230KV supply to the Gordonsville line)

SAFETY EVALUATION SUMMARY

A power supply is required for a STREAMLINER trailer which will be used to pump oil from transformer #3. Power will be fed through the construction power supply tie in from 34.5 KV bus #4 to 34.5 KV bus #5. From there, power will be backfed to the low side of Transformer #3. Power will be stepped down and tapped off as required to power the Streamliner.

SAFETY EVALUATION NUMBER 93-SE-OT-070

DESCRIPTION

1(2)-IPM-FW-V-001, Main Feedwater Regulating Valve Inspection

These procedures install a temporary modification that bypasses an ESF function; namely, feedwater isolation.

SAFETY EVALUATION SUMMARY

This procedure is performed as a part of maintenance on the main feedwater reg valves. The procedure may be performed in modes 3, 4, 5, or 6, but the feedwater isolation ESF signal is required to be operable in modes 3 and 4. The procedure insures that an alternate method of isolation is in place prior to installing the jumper, if it is required. If no feedwater isolation signal is present or required, the procedure does not install the jumper.

SAFETY EVALUATION NUMBER 93-SE-OT-071

DESCRIPTION

Position paper "OPNS-1407 Position for Multiple Instrument Failures" which states that during situations of multiple instrument failures of vital instrumentation for a specified parameter, Tech Spec 3.0.3 should be entered vice reactor trip or ESF actuation. This is provided that it can be verified that the reactor is in a safe operating condition.

SAFETY EVALUATION SUMMARY

The major item considered was whether or not a failure of the reactor to trip or an ESF actuation to initiate upon the coincidence being met due to instrument failures, was cause for the reactor to be tripped manually. Review of the bases for Tech Spec 3.0.3 concluded that it would be appropriate to enter this action based on two examples given in the bases. In addition this position does not conflict with 10CFR50.62 Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light water cooled nuclear power plants. This position does not constitute an unreviewed safety question since it clarifies the use of an existing Tech Spec in a manner consistent with the Bases. In addition, implementation of this position will reduce the transients (i.e. reactor trips) on the unit. Entry into Tech Spec 3.0.3 would only be performed if the unit was verified to be in a safe condition and the problem was with the instrumentation.

SAFETY EVALUATION NUMBER 93-SE-OT-072

DESCRIPTION

Evaluation to allow use of ethanolamine, instead of morpholine, for secondary system pH and corrosion control.

SAFETY EVALUATION SUMMARY

Ethanolamine is a more efficient corrosion control agent than morpholine in the two phase regions because it can provide a higher liquid phase Ph in those areas. It is also a stronger base than morpholine and requires a lower concentration to achieve the same chemistry conditions as morpholine. This translates to lower loading of polishers and polisher run times should increase. Use of ethanolamine is endorsed by EPRI and Westinghouse. Testing has shown that the ethanolamine is not detrimental to any gasket material, ion exchange resins or other materials (such as Buna-N, viton A, EPDM, etc) that are used in secondary systems. Use of the ethanolamine at other plants has netted a decrease in overall iron transport which results in less sludge in the steam generators.

SAFETY EVALUATION NUMBER 93-SE-OT-073

DESCRIPTION

Testing of the SI Accumulator discharge check valves to ensure they will fully open.

SAFETY EVALUATION SUMMARY

The test involves flowing each accumulator to the core to verify that the discharge check valves fully open. The test will be performed with fuel removed, the upper internals installed and the reactor head off. All equipment and systems will be operated within design limits, which precludes failure of the RCS piping. The test will be performed such that nitrogen gas from the accumulator will not be introduced into the RCS. Reactivity addition is precluded by ensuring accumulator boron concentration is greater than that of the RCS.

SAFETY EVALUATION NUMBER 93-SE-OT-074

DESCRIPTION

Tech Spec change request #300 was written to delete the lists in unit 2 Tech Specs dealing with thermal overloads, normally de-energized power circuits and containment penetration overcurrent protection devices. Also the list of containment isolation valves will be deleted from both units Tech Specs.

SAFETY EVALUATION SUMMARY

These lists will be relocated to station controlled procedures and is in accordance with guidance provided in NRC Generic Letter 91-08. The Limiting Conditions for Operation, Action Statements, and Surveillance Requirements will still apply to all components in the lists.

SAFETY EVALUATION NUMBER 93-SE-OT-075

DESCRIPTION

Unit 2 Core Reload 10 fuel assemblies will have the following mechanical features: 1) vibration suppression damping assemblies placed in Vantage 5H fuel assemblies used in baffle locations, 2)rotation of alternate mixing vane grids for fresh fuel to suppress assembly vibration, and 3) minor modifications to fresh assemblies to enhance debris resistance.

SAFETY EVALUATION SUMMARY

Applicable safety analyses remain bounded with the mechanical features assuming the reduced minimum measured RCS flow Technical Specification change. The effects of the reload parameter variations were accommodated within the conservatism of the assumptions used in the applicable safety analyses.

SAFETY EVALUATION NUMBER 93-SE-OT-076

DESCRIPTION

Provides engineering analysis of the acceptability of a design deficiency identified with the Unit 1 RCP UF relay 125 VDC power, that is the neutral cable is not color designated and consequently routing of the DC power supply cables to the UF auxiliary relay panels is not properly separated consistent with the design basis in the UFSAR.

SAFETY EVALUATION SUMMARY

Existing design is acceptable based on no credible single failures having been identified which compromise the ultimate initiation of a reactor trip for an underfrequency event. The evaluation was conducted per IEEE Standard 379-1977 which provides guidance for assuring that single failure criterion is not violated where independence cannot be readily demonstrated.

SAFETY EVALUATION NUMBER 93-SE-OT-077

DESCRIPTION

This Safety Evaluation is essentially a revision of 93-SE-OT-075 based on two assemblies scheduled for reuse having been found to contain failed fuel. Thus, Unit 2 Core Reload 10 was redesigned. Unit 2 Core Reload 10 fuel assemblies will have the following mechanical features: 1) vibration suppression damping assemblies placed in Vantage 5H fuel assemblies used in baffle locations, 2) rotation of alternate mixing vane grids for fresh fuel to suppress assembly vibration, and 3) minor modifications to fresh assemblies to enhance debris resistance.

SAFETY EVALUATION SUMMARY

Applicable safety analyses remain bounded with the mechanical features assuming the reduced minimum measured RCS flow Technical Specification change. The effects of the reload parameter variations were accommodated within the conservatism of the assumptions used in the applicable safety analyses.

SAFETY EVALUATION NUMBER 93-SE-OT-078

DESCRIPTION

Changes the UFSAR to clarify the purpose of the Unit 1 Motor Driven Auxiliary Feedwater Pumps discharge pressure control valves and document the maximum flows that the Unit 1 Motor Driven Auxiliary Feedwater Pumps can safely maintain for 30 minutes without cavitating.

SAFETY EVALUATION SUMMARY

Based on engineering analysis, the design basis of the MDAFW Pumps to deliver at least 600 gpm to the Steam Generators at a discharge pressure of at least 900 psig is met.

SAFETY EVALUATION NUMBER 93-SE-OT-079

DESCRIPTION

The purpose of this test is to stroke the BIT inlet and Hot Leg Normal and Alternate Safety Injection isolation valves against a delta-p, record VOTES data, and satisfy the intent of Generic Letter 89-10.

SAFETY EVALUATION SUMMARY

Uncontrolled boron dilution is not credible with the plant in mode 5 or 6 as the primary grade water flowpath is procedurally and administratively blocked and this test does not manipulate that flowpath. With regard to LBLOCA, the Charging system will be operable but SI will not be required and the solid state protection fuses will be pulled prior to this test. The valves themselves will be operated as designed and in accordance with approved procedures and the UFSAR.

SAFETY EVALUATION NUMBER 93-SE-OT-080

DESCRIPTION

New procedure provides a means to compensate for the failure of a single hot leg RTD and its spare, changing the T-hot-average calculation from averaging 3 RTD's to averaging 2 RTD's in the failed loop T-hot summator.

SAFETY EVALUATION SUMMARY

The change will not affect the time response of the system. Since the protection system is designed as two out of three loops, even if the affected loop were to fail, it would not disable the protection system. Thus the RPS Overpower and Overtemperature Delta-T protection functions are maintained within the design basis.

SAFETY EVALUATION NUMBER 93-SE-OT-081

DESCRIPTION

This Safety Evaluation is essentially a revision of 93-SE-OT-077 adding an evaluation of an extension of the LBLOCA S/G tube plugging level from 20 percent to 23.5 percent plugging to bound the current "C" S/G plugging level of 23.11 percent. Unit 2 Core Reload 10 fuel assemblies will have the following mechanical features: 1) vibration suppression damping assemblies placed in Vantage 5H fuel assemblies used in baffle locations, 2) rotation of alternate mixing vane grids for fresh fuel to suppress assembly vibration, and 3) minor modifications to fresh assemblies to enhance debris resistance.

SAFETY EVALUATION SUMMARY

Applicable safety analyses remain bounded with the mechanical features assuming the reduced minimum measured RCS flow Technical Specification change. The effects of the reload parameter variations were accommodated within the conservatism of the assumptions used in the applicable safety analyses.

SAFETY EVALUATION NUMBER 93-SE-OT-082

DESCRIPTION

Use freeze seals to isolate the Unit 1 Cation Bed Demineralzer upstream of its inlet isolation valve and downstream of its outlet isolation valve in order to isolate the entirety from the Letdown system and allow the valves to be rebuilt as they are leaking by before 01-CH-4, inlet to IX and after 01-CH-10, outlet of IX so these valves can be rebuilt.

SAFETY EVALUATION SUMMARY

The work is corrective in nature and no modifications are involved. The design basis of the Letdown system is not changed. The Cation Bed is not required to be in service at all times. The contingency plans established for Operations and Maintenance are adequate if a freeze seal fails.

SAFETY EVALUATION NUMBER 93-SE-OT-083

DESCRIPTION

Technical Specification 5.3.1 for both Unit 1 and 2 is being modified to allow the use of solid filler rods of stainless steel or zirconium alloy in place of failed fuel rods. The individual fuel rod uranium weight limit is also being deleted.

SAFETY EVALUATION SUMMARY

The impact of the use of filler rods on the mechanical, neutronic, thermal-hydraulic, and safety analyses (both LOCA and non-LOCA) was assessed. The fuel vendor evaluated a replacement of up to a 25 percent of fuel rods in a fuel assembly as meeting mechanical requirements. Neutronic and thermal-hydraulic aspects will in general be bounded by full fuel rod assemblies. Cycle specific reload evaluations will be conducted. Fuel rod weight does not have any direct bearing on fuel performance, power limits, power operating level, or decay heat rate, which will continue to be bound by existing analyses and Technical Specification limits.

SAFETY EVALUATION NUMBER 93-SE-OT-084

DESCRIPTION

This Safety Evaluation essentially cover sheets the referenced Westinghouse Safety Evaluation which approves the use of sleeved cable stabilizers, stacked cable stabilizers, and sentinel plugs to surround unstabilized S/G tubes which have circumferential indications, thus preventing interaction with neighboring tubes and preventing tube rupture.

SAFETY EVALUATION SUMMARY

Per Westinghouse, the installation of the tube stabilizers and sentinel plugs does not constitue a change to the plant as described in the FSAR, it does not impact the probability or consequences of an accident, nor does it decrease any margin of safety.

SAFETY EVALUATION NUMBER 93-SE-OT-085

DESCRIPTION

Change the Technical Specification surveillance frequency of the AFW pumps from monthly to a staggered quarterly basis, include a 3.0.4 exclusion for the TDAFWP for startup and modify the startup surveillance requirements to provide a 24-hour time limit to test the TDAFWP, establish an Action Statement for the steam supply lines for the TDAFWP, clarify the Action Statement for three inoperable AFWP's, change the surveillance requirements for the AFW flow paths including a shutdown of greater than 30 days stipulation, update the Technical Specification Bases section for the AFW system, and revise the UFSAR to reflect these changes. Purpose is to reduce the testing of the pumps at power.

SAFETY EVALUATION SUMMARY

Testing the AFW system on a quarterly basis is adequate to ensure that the system is capable of performing its design function. The method of testing will remain the same. Therefore, accident analyses remain bounded and no margin to safety is reduced.

SAFETY EVALUATION NUMBER 93-SE-OT-086

DESCRIPTION

Delete the surveillance requirement of Technical Specification 4.5.2.h.1.c to use a value of at least 48.3 gpm as a simulated seal injection flow during charging pump flow testing per Technical Specification 4.5.2. The requirement is excessively restrictive and does not take into account differences in HHSI pump performance.

SAFETY EVALUATION SUMMARY

The requirements for the minimum total of the two lowest SI branch flows and the maximum total HHSI pump flowrate will continue to be met ensuring flows are balanced and meet the safety analyses.

SAFETY EVALUATION NUMBER 93-SE-OT-087

DESCRIPTION

Use TIP/CECOR, a Virginia Power version of the CECOR code, to infer a three-dimensional core power distribution from a limited number of moveable incore detector measurements. This flux map analysis is currently performed using the INCORE code.

SAFETY EVALUATION SUMMARY

TIP/CORE offers improved accuracy due to better axial representation of inputs which will improve consistency in core power distribution monitoring/surveillance and core follow. The extensive qualifying of TIP/CORE has demonstrated that it is essentially an equivalent software replacement for INCORE. Core limits will not be changed.

SAFETY EVALUATION NUMBER 93-SE-OT-088

DESCRIPTION

Technical Specifications are being revised to note that 1) SNSOC will only review new procedures and procedure changes that require a safety evaluation, and 2) the MSRC will only review a sample of safety evaluations and SNSOC meeting minutes and reports.

SAFETY EVALUATION SUMMARY

No modifications to the plant are being made by this change. Activities important to nuclear safety will still be reviewed by SNSOC and the adequacy and effectiveness of SNSOC reviews will still be assured by the MSRC's overview.

SAFETY EVALUATION NUMBER 93-SE-OT-089

DESCRIPTION

This Safety Evaluation is a revision of 93-SE-OT-85 splitting out the quarterly testing change from the other issues. Change the Technical Specification surveillance frequency of the AFW pumps from monthly to a staggered quarterly basis and revise the UFSAR to reflect this change. Purpose is to reduce the testing of the pumps at power.

SAFETY EVALUATION SUMMARY

Testing the AFW system on a quarterly basis is adequate to ensure that the system is capable of performing its design function. The method of testing will remain the same. Therefore, accident analyses remain bounded and no margin to safety is reduced.

SAFETY EVALUATION NUMBER 93-SE-OT-090

DESCRIPTION

UFSAR Table 5.2-25 UFSAR Section 9.3.4.2.4.7

This SE is for a change to the UFSAR. On Table 5.2-25, delete references to RCS temperature and add notes to allow the RCS Hydrogen concentration to go as low as 15 cc/kg within 24 hours prior to shutdown and to allow RCS Hydrogen concentration to range from 15 to 50 cc/kg 24 hours following reactor criticality. In Section 9.3.4.2.7, delete the use of specific values of Hydrogen concentration and leave in the general terms "required equilibrium concentration."

SAFETY EVALUATION SUMMARY

This new Hydrogen concentration requirement will prevent delays in unit startup while still meeting the purpose of the requirement which is to scavange the oxidizing species formed by radiolysis and minimize the corrosion of primary system materials (see Technical Justification for Surry Unit 1 RCS Hydrogen Concentration for Critticality memo from W. A. Thorton dated 5/1/92 which is based on Westinghouse recommendations). It will also facilitate hydrogen degassification following shutdown, therby preventing contamination of refueling water and containment atmosphere. This SE supports changing the UFSAR.

SAFETY EVALUATION NUMBER 93-SE-OT-091

DESCRIPTION

UFSAR change to revise the Chemical Addition Tank (CAT) chloride concentration from <0.15 ppm to <2000 ppm an section 6.2.2.3.

SAFETY EVALUATION SUMMARY

Due to the nature of sodium hydroxide, there is no viable method to maintain the CAT at <0.15 ppm. The limit of 2000 ppm has been analyzed by engineering and is fully acceptable. The effect of slightly higher than expected concentrations of chlorides in the sump following a DBA will have minimal impact on ESF systems.

SAFETY EVALUATION NUMBER 93-SE-OT-092

DESCRIPTION

Revision to OP-51.1 (Component Cooling System) to allow for cross connect of unit 1 and unit 2 CC systems.

SAFETY EVALUATION SUMMARY

The CC system was not designed to mitigate any design basis accident. Normal operation of the CC system cross connected will distribute the CC heat load on the CC heat exchangers more evenly enabling them to operate more efficiently. During hot weather, an additional SW pump to cool the unit with CC common loads may no longer be required thus saving energy. Reliability would be increased since the standby pump from either unit would be available to operate as needed in support of both units.

SAFETY EVALUATION NUMBER 93-SE-OT-093

DESCRIPTION

Installation of a stem bushing collar on the "A" governor valve (01-MS-GOV-1A).

SAFETY EVALUATION SUMMARY

The bushing is being installed to alleviate possible gasket compressibility concerns brought up by Westinghouse. The collar will prevent the stem bushing from falling into the steam flow stream in the event that the gasket fails. Installation of the collar will not affect operation of the governor valve in any way. Following installation of the collar, freedom of movement will be verified.

SAFETY EVALUATION NUMBER 93-SE-OT-094

DESCRIPTION

Approval of Nuclear Plant Chemistry Manual (NPCM) and revision of UFSAR to incorporate new chemistry parameter values.

SAFETY EVALUATION SUMMARY

The new chemistry parameters were developed based on Tech Specs, UFSAR, and/or industry guidelines (EPRI and Westinghouse standards etc.). The new chemistry parameters are equal to or more restrictive than those currently in the UFSAR.

SAFETY EVALUATION NUMBER 93-SE-OT-095

DESCRIPTION

North Anna's Emergency Plan Implementing procedures are being revised to change the calculation and estimation of worker and population dose rates from R.G. 1.109 to ICRP-26 and EPA-400 guidelines and methodology. This also changes terminology from whole body to TEDE and thyroid dose to CDE.

SAFETY EVALUATION SUMMARY

The adoption of EPA-400 guidelines provides for a more complete estimation of potential population dose and lower protective action guidelines which could reduce the total dose to the general public under accident scenarios.

SAFETY EVALUATION NUMBER 93-SE-OT-096

DESCRIPTION

Revision to the Operational Quality Assurance Program Topical Report which transfers the administration of the Internal Audit Program from the Manager-Quality Assurance (Corporate) to the Manager-Quality Assurance (Station).

SAFETY EVALUATION SUMMARY

The proposed change only moves the responsibility for the administration of Quality Assurance's Internal Audit Program, not the QA External Audit Program of vendors and contractors or the independent biennial internal audit of activities required to meet the criteria of Appendix "B" 10CFR50. This administrative change affects the auditing process only. There is no affect to any systems or equipment.

SAFETY EVALUATION NUMBER 93-SE-OT-097

DESCRIPTION

0-PT-75.18 provides instructions for performance of delta-P testing of various SW MOVs in response to NRC Generic Letter 89-10.

SAFETY EVALUATION SUMMARY

The SW MOVs will be tested to ensure that they will operate under design basis conditions. The testing will be within the design limits of the system and individual components.

SAFETY EVALUATION NUMBER 93-SE-OT-098

DESCRIPTION

Revision to UFSAR section 16.2.1.2.1. The revision revises the surveillance requirement (16.2.1.2.1.1) to perform a flush of the fire suppression water system at least once per 12 months.

SAFETY EVALUATION SUMMARY

There are no changes to the system design or operation. The existing surveillance requirement stated to flush the system as necessary to maintain water chemistry within acceptable limits. The fire protection system water supply is from Lake Anna. The station did is not required to maintain water chemistry of the lake. The basis for the surveillance was to reduce internal tuberculation in the unlined piping since it would reduce flow through the pipe. The FP piping at NAPS is cement lined therefore the potential for tuberculation is not as significant.

1993 50.59 SAFETY EVALUATIONS REPORTABLE TO THE NRC

ENGINEERING WORK REQUESTS (EWR)

89-SE-MOD-083
89-SE-MOD-102
89-SE-MOD-147
90-SE-MOD-035
90-SE-MOD-038
90-SE-MOD-133
90-SE-MOD-148
90-SE-MOD-166
91-SE-MOD-042
91-SE-MOD-054
93-SE-MOD-079

ENGINEERING WORK REQUEST 89-268K
LIMIT SWITCH SETPOINT CHANGE FOR RSHx INLET MOVS
NORTH ANNA UNIT 1

DESCRIPTION

This EWR is the vehicle for changing the limit switch setpoints for 2-SW-MOV-203C&D; changed as a result of 2-PT-75.6. The A&B valves remain unchanged from the last time the PT was run. The C valve was changed to 55% open and the D valve was changed to 38.2 percent open.

SUMMARY OF SAFETY ANALYSIS (89-SE-MOD-~~102~~ 083)

This EWR did not create an unreviewed safety question as defined in 10 CFR 50.59.

The PT (2-PT-75.6) balances RSHx flow to conform to the Design Basis. Having done that the NASD, by procedure, is changed. This method was previously evaluated and approved by SNSOC. As such, there are no unreviewed safety questions.

ENGINEERING WORK REQUEST 89-216,B,C,E,F,H,I,J,K
RADIATION MONITOR SWITCH REPLACEMENT
NORTH ANNA UNIT 1

DESCRIPTION

This modification changes the wiring of the transfe: ontrol circuit in the compartment of 1-RM-P-159A&B. The modification changed the pump selector switch wiring so that the switch switches 120V instead of 480V. This modification involved the addition of two breakers so that one pump may be worked on while the other remained energized.

SUMMARY OF SAFETY ANALYSIS (89-SE-MOD-102)

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

The modification does not affect the ability of the containment gaseous and particulate Rad Monitors to perform their RCS leakage monitoring function and containment purge and exhaust isolation system funcation as described in the UFSAR.

EWR 88-317 A, B

MODIFICATION TO RAD MONITOR ENCLOSURE BOX
(SE #89-SE-MOD-147)

DESCRIPTION

The EWR modified the mounting bracket inside the radiation detector shield box to allow faster disassembly and reassembly of the detector. The modification only revised the mounting bracket detail and was installed for additional radiation monitors in the plant (Addendum A and B allowed additional locations).

SUMMARY OF SAFETY ANALYSIS

The original safety evaluation was utilized for addendums A and B. The safety evaluation concluded that an unreviewed safety question does not exist for the bracket improvement.

EWR 90-027 and Addenda A-K

BORON INJECTION TANK PRESSURE TRANSMITTER REMOVAL
NORTH ANNA UNIT 1 & 2

DESCRIPTION

The boron injection tank pressure transmitters were located in high radiation areas with airborne radioactivity and were repeatedly requiring maintenance. The transmitters were nonsafety related and were not required for any emergency procedures. The transmitters for both unit #1 and 2 were removed along with associated piping, heat tracing and loop components. Although the transmitters were nonsafety related, the UFSAR referenced them and required revision.

SUMMARY OF SAFETY ANALYSIS (90-SE-MOD-035)

This transmitter removal did not create an unreviewed safety question as defined by 10CFR50.59.

A. The transmitter removal did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

The transmitters and associated equipment had no control function and were used for information only. The transmitters were isolated from the operating systems prior to removal so that no operational systems or equipment were affected.

B. The implementation of this modification did not create a possibility for an accident of a different type than previously evaluated in the Final Safety Analysis Report.

The transmitters were isolated from the operating system and removed. All mechanical and electrical isolations were performed in accordance with all applicable codes and standards so that the function and reliability of the operating systems was not affected.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

The transmitters and associated equipment are not addressed in any Technical Specification basis.

EWR 87-646, D ,E, F

REPLACEMENT OF RHR MONORAIL
NORTH ANNA UNIT 1

DESCRIPTION

The major issues considered were structural seismic qualification and adherence to the ALARA program. This proposed change re-routes an existing monorail. The monorail re-route, including supports and bolts have been seismically evaluated and qualified in SWEC Calculation No. 14938.76-S-1. The reason for this proposed change is the reduction of personnel radiation exposure. The existing route requires that a shield wall be removed so the RHR pump can be transported to an area of decreased radiation level. The need to remove the shield is being eliminated this reducing personal exposure.

SUMMARY OF SAFETY ANALYSIS - (90-SE-MOD-038)

An unreviewed safety question does not exist because the existing monorail is seismically qualified as is the re-routed version. The change should be allowed because it will reduce man-rem exposure levels (long term) and it creates no additional safety concerns to the facility nor does it increase the risk of radiological exposure to the general public.

EWR 89-314A - E

REMOVAL OF METAL IMPACT MONITORING SYSTEM (MIMS) AND RECONFIGURATION OF THE VIBRATION AND LOOSE PARTS MONITORING SYSTEM (VLPMS) NORTH ANNA UNITS 1 & 2

DESCRIPTION

The MIMS was installed in the control room as a temporary monitoring system 11 years ago to provide additional surveillance of certain thermal sleeves in the RCS. These certain thermal sleeves has a past history of cracking due to high and low cycle fatigue. The MIMS utilized the passive accelerometer wiring of the existing VLPMS to access several junction boxes in reactor containment. From these junction boxes, temporary cable was installed to locations of temporarily installed accelerometers on the hot and cold legs of the RCS.

Westinghouse performed a Loose Parts Monitoring System Thermal Sleeve Evaluation (WCAP-12355), August 1989 on units 1 and 2. The study concluded the MIMS nor the VLPMS is no longer required for thermal sleeve surveillance because the remaining thermal sleeves will not fail due to low or high cycle fatigue.

The MIMS panel was removed from the control room, the temporary cabling from the junction boxes to the temporarily installed accelerometers on the hot and cold legs of the RCS inside reactor containment were removed, and the passive ports of the VLPMS were restored to their original configuration.

SUMMARY OF SAFETY ANALYSIS (90-SE-MOD-133)

The EWR did not create an unreviewed safety question as defined in 10 CFR 50.59.

A. The implementation of this modification did not increase the probability of occurrence or consequences of an accident or malfunctions of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

   The MIMS was installed only as a temporary system to strictly monitor the thermal sleeves in question. Westinghouse has performed a study to evaluate the thermal sleeves in question in the RCS. The investigation conclude that the MIMS is no longer required the thermal sleeves because they will remain intact through low and high cycle fatigue tests. The VLPMS will continue to monitor the RCS for loose parts.

B. The implementation of the this modificatio, die not create a possibility for an accident or a malfunction of a different type that any previously evaluated in the Final Safety Analysis Report.

The Westinghouse study concluded that the performance of the thermal sleeves will not be reduced due to low and high cycle fatigue. Removal of the MIMS will not affect any other system will restore the passive ports of the VLPMS to their original configuration.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

The design bases of the VLPMS are preserved. No credit is taken for the MIMS in any Safety Analysis or Technical Specification safety margin. Radiography of remaining thermal sleeves will periodically performed.

EWR 90-327, A, B, C

REPLACE CONCRETE LIFTING INSERTS
(SE #90-SE-MOD-148)

DESCRIPTION

Concrete inserts used for rigging out the removable concrete slabs degraded from corrosion. New lifting inserts were installed in concrete slabs for the Unit 1 and 2 Main Steam Valve House and the Auxiliary Service Water Pump house located at the intake structure. Stainless steel Drillco Maxibolts were utilized for the superior corrosion resistance and rated load capacity.

SUMMARY OF SAFETY ANALYSIS

The safety evaluation considered the impact of failure of the insert while lifting the concrete slabs. The safety evaluation concluded that an unreviewed safety question does not exist.

EWR 90-351 and Addenda A-C

SERVICE WATER HEADER TRANSFER PUMP SUPPORTS
NORTH ANNA UNIT 1

DESCRIPTION

The service water header transfer pump allows the draining of the supply and discharge headers. This pump was originally a temporary installation. The pump and associated piping were evaluated as a permanent installation. The pump and associated piping were determined to be non safety related up to the isolation valve at the header which is closed during normal plant operation. However, supports were added to at the isolation valve to comply with the seismic requirements of the service water system.

SUMMARY OF SAFETY ANALYSIS (90-SE-MOD-166)

The addition of the seismic supports did not create an unreviewed safety question as defined by 10CFR50.59.

A. The non safety related piping configuration which is attached to the safety related isolation valve did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

   The isolation valve is closed during plant operation and the addition of supports ensure that the valve is not affected by a seismic event. Therefore, the integrity of the valve as a system pressure boundary was maintained.

B. The implementation of this modification did not create a possibility for an accident of a different type than previously evaluated in the Final Safety Analysis Report.

   The isolation valve is normally closed and is seismically supported so that the non safety related piping did not affect the function, operation or reliability of the safety related service water system.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

   The non safety related pipe and the isolation valve are not addressed in any Technical Specification basis.

EWR 89-318A - E

INSTALLATION OF FIVE VALVE MANIFOLD ON LEVEL TRANSMITTERS 01-RC-LT-1000 AND 02-RC-LT-2000 NORTH ANNA UNITS 1 & 2

DESCRIPTION

Level Transmitters (01-RC-LT-1000 and 02-RC-LT-2000) did not have an isolation valve or calibration tap on the low side of the transmitter reference leg. New calculation EE-0079, Rev 1 requires the level transmitter be calibrated with an input on the low side. This requires a complicated valve line up and filling.

A five valve manifold arrangement and a new level transmitter were installed for level transmitters (01-RC-LT-1000 and 02-RC-LT-2000). The unnecessary calibration pot, sealed reference, and some associated tubing has been removed.

SUMMARY OF SAFETY ANALYSIS (91-SE-MOD-042)

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

A. The implementation of this modification did not increase the probability of occurrence or consequences of an accident or malfunctions of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

The installation of the new five valve manifold arrangement and level transmitters have not altered the operation of the pressurizer level transmitters (01-RC-LT-1000 and 02-RC-LT-2000) and have not altered the operation of the pressurizer or the RCS.

B. The implementation of this modification did not create a possibility for an accident or a malfunction of a different type than any previously evaluated in the Final Safety Analysis Report.

The installation of the new five valve manifold arrangement and level transmitters have not increased the chances of accidental depressurization of the RCS. The new five valve manifold arrangement and level transmitter have been tested in accordance with approved specifications.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

The design bases of the RCS have been preserved. The installation of the new five valve manifold arrangement and level transmitters have not altered the operation of the pressurizer level transmitters (01-RC-LT-1000 and 02-RC-LT-2000). Thus, the operation of the pressurizer or the RCS has not been affected.

EWR 90-385,A

Removal of Cation Columns and CO_2 Scrubbers
NAPS - Units 1 & 2

DESCRIPTION

The Condensate Polishing System was originally supplied with equipment which allowed for the conductivity to be continuously monitored using a cation column with a conductivity cell. This equipment was no longer being used as more accurate portable equipment had become available. Therefore this outdated, unused equipment has removed by EWR 90-385. Although the Condensate Polishing System is non safety related, the monitoring equipment removed was described in the UFSAR, thus a UFSAR Change Request and Safety Evaluation were required.

SUMMARY OF SAFETY ANALYSIS (91-SE-MOD-054)

The Condensate Polishing System and associated conductivity sampling equipment is non safety related. The removal of the unused conductivity monitoring equipment does not alter the function of the condensate polishing system. The conductivity is still being monitored, only portable more accurate equipment is no being used. The UFSAR was revised to reflect this change.

Since this modification did not change the function of the Condensate Polishing System and no safety related equipment was affected, there was no increase to the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the UFSAR. No new accidents were created as a result of this modification as the condensate polishing system conductivity is still monitored. The margin of safety was not affected. Therefore an unreviewed safety question does not exist.

EWR 93-009
Safety Evaluation #93-SE-MOD-079

Revise Pressure Rating of Aux Feedwater Discharge Piping
NAPS - Units 1 & 2

DESCRIPTION

The design temperature and pressure for the turbine driven aux feedwater pump was rerated by this EWR. This was done to allow the relief valves on the discharge of the pumps to be set at a higher setpoint. Per original piping code requirements, relief valve set pressure is to be the same as system design pressure.

SUMMARY OF SAFETY ANALYSIS

The aux feedwater system provides water to the steam generators at times when the normal feedwater system is not available. In accident conditions, aux feedwater maintains the heat sink capabilities of the steam generators. All accidents were reviewed and the aux feedwater system is required for all accidents which have a safety injection actuation. Accidental depressurization of the main steam system, a small break LOCA, major secondary system pipe rupture, steam generator tube rupture, main steam line break and large break LOCA were considered to be applicable.

Accident probability has not been increased as the aux feedwater system is for accident mitigation purposes. The system is a backup for normal feedwater and does not contribute to the probability of occurrence of an accident.

The consequences of any of these accidents are not affected. The reduced temperature rating of the pipe is within the originally evaluated temperature range for ability to remove decay heat from the steam generator. The increased pressure rating is within the capability of the pipe. The system will still be operated within the original operating temperatures and pressures and will still function as designed.

No unique accident probabilities are created. The function, operation and performance of the TDAFWP and the aux feedwater system is not changing.

Margin of Safety is maintained because the integrity and reliability of the aux feedwater system has been evaluated and is not affected. The reduction of design feedwater temperature is below the value assumed to allow for decay heat removal from the steam generators. This temperature reduction is in the conservative direction as it allows for more heat removal than the maximum temperature (120°F) assumed by the UFSAR.

EWR 91-409A

REVISION TO N.A.S.D AND UFSAR
NORTH ANNA / UNITS 1 & 2

DESCRIPTION

During the Electrical Distribution System Functional Inspection (EDSFI) conducted at North Anna from July 29,1991 through August 30, 1991 discrepancies were noted between the Emergency Diesel Generators (EDG) setpoint document and Colt Industries Service Manual for the EDG and UFSAR. As a result of these discrepancies, the North Anna Setpoint Document, UFSAR and the Training Manual were revised.

The following setpoints were revised:

Diesel Lube Oil Filter High Differential Pressure
Lube Oil Low Temperature Alarm
Jacket Coolant Low Pressure

The UFSAR was revised to reflect the proper maximum ambient air temperature and EDG radiator inlet air temperature. The UFSAR was also changed to reflect the correct value of the EDG room fan capacity.

SUMMARY OF SAFETY ANALYSIS

This modification does not constitute an "unreviewed safety question" as defined in 10CFR50.59.

a. The implementation of this modification did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

   This activity changed two setpoints to agree with the instructions found in the Colt Industries Service Manual. Other changes involved document clarification. There is no change in the probability of an accident. Operation and function of the EDG in a Design Basis Accident is not affected.

b. The implementation of this modification did not create a possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report.

This activity dealt with EDG setpoints and documentation. The probability of any type of accident is not increased. EDG vendor documentation provides justification for new setpoints since operation of the EDG is not affected. No new type of accident is introduced.

c. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

This change is a clarification to the EDG documentation in the UFSAR and NCRODP-55. In addition, this activity changed the EDG setpoints to comply with the vendors service manual. Operability of the EDG's is maintained therefore the margin of safety as described in Tech Specs is not affected.

EWR 90-319E

REPLACEMENT OF 1-SW-RV-101A AND 1-SW-RV-101B RELIEF VALVES NORTH ANNA UNIT 1

DESCRIPTION

1-SW-RV-101A and 1-SW-RV-101B were found with no manufacturers' nameplate. Per the ASME pressure vessel code, this was not acceptable and therefore required replacement. The existing valves were Farris type 2740 relief valves. The valves were replaced with type 2740-ULR/S4 valves to improve component reliability.

SUMMARY OF SAFETY ANALYSIS

This relief valve replacement does not constitute an unreviewed safety question or require a modification to the Technical Specifications. The function and operation of the relief valve and the system in which it is installed remains the same. The replacement does not increase the probability, consequence or possibility of an accident. The Margin of Safety as set forth in the Tech Specs. is not affected. Accidents of a different type than previously analyzed are not possible.

EWR 90-319D

REPLACEMENT OF 1-CC-RV-102 AND 1-CC-RV-104B RELIEF VALVES NORTH ANNA UNIT 1

DESCRIPTION

1-CC-RV-102 and 1-CC-RV-104B were Farris type 2740 relief valves which required replacement. Exact replacements are no longer manufactured. Therefore, the valves were replaced with upgraded type 2740-ULR/S4 valves to improve component reliability.

SUMMARY OF SAFETY ANALYSIS

This relief valve replacement does not constitute an unreviewed safety question or require a modification to the Technical Specifications. The function and operation of the relief valve and the system in which it is installed remains the same. The replacement does not increase the probability, consequence or possibility of an accident. The Margin of Safety as set forth in the Tech Specs. is not affected. Accidents of a different type than previously analyzed are not possible.

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TEMPORARY MODIFICATIONS/JUMPERS

90-SE-JMP-033 Rev 1
93-SE-JMP-001
93-SE-JMP-002
93-SE-JMP-003
93-SE-JMP-004
93-SE-JMP-005
93-SE-JMP-006
93-SE-JMP-007
93-SE-JMP-008
93-SE-JMP-009
93-SE-JMP-010
93-SE-JMP-011
93-SE-JMP-012
93-SE-JMP-013
93-SE-JMP-014
93-SE-JMP-015
93-SE-JMP-016
93-SE-JMP-017
93-SE-JMP-018
93-SE-JMP-019
93-SE-JMP-020
93-SE-JMP-021
93-SE-JMP-022
93-SE-JMP-023
93-SE-JMP-024
93-SE-JMP-025
93-SE-JMP-026

SAFETY EVALUATION NUMBER 90-SE-JMP-033 Rev 1

DESCRIPTION

Add temporary ventilation to the control rod drive rooms to provide additional cooling to the pressurizer heater breaker panel and fuse boxes.

SAFETY EVALUATION SUMMARY

Temporary ventilation was added to the control rod drive rooms that was seismically restrained and powered from a non-emergency supply. Proper ventilation and control room differential pressure indication was verified.

SAFETY EVALUATION NUMBER 93-SE-JMP-001

DESCRIPTION

Jumper N1-1558
A manual isolation valve(s) will be installed on the polar crane bridge brake hydraulic cylinder(s). The valve(s) will be closed and thus isolate the hydraulic bleeding solenoid(s) from the hydraulic cylinder(s).

SAFETY EVALUATION SUMMARY

One train of the polar crane brake system cannot maintain hydraulic pressure. A troubleshooting procedure was developed which will test the master cylinder, the hydraulic cylinders, and the hydraulic bleed SOV. There are two independent brake systems on the polar crane each utilizing two brakes. Each brake has its own hydraulic cylinder and bleed SOV. In order to test each SOV independently the manual isolation valve will be installed. If one SOV leaks by it will be replaced with a spare from stock. If both SOVs are bad the isolation valves will remain in place since there is only one spare SOV in stock. Once the spares are received they will be replaced and the jumper removed.

SAFETY EVALUATION NUMBER 93-SE-JMP-002

DESCRIPTION

Jumper N1-1559
The underload interlock of the manipulator crane will be bypassed via an electrical jumper.

SAFETY EVALUATION SUMMARY

The original Dillon load cell for the manipulator crane cannot be located. A replacement load cell is available, but it lacks the underload protection interlock capacity. This interlock will be jumpered out in order to use the replacement.

SAFETY EVALUATION NUMBER 93-SE-JMP-003

DESCRIPTION

Jumper N1-1559
The vent path for the PDTT to the VG header has been isolated to prevent potential radiological concerns from Unit 2 PDTT vents entering Unit 1 containment during the Unit 1 outage.

SAFETY EVALUATION SUMMARY

This document evaluates the temporary installation of a PDTT vent. This vent will consist of a tygon tubing arrangement to the containment exhaust purge duct originating from 01-DG-106. The purpose of this jumper is to provide a method for venting the Unit 1 PDTT during the SGR outage since the vent path to the VG header has been isolated.

SAFETY EVALUATION NUMBER 93-SE-JMP-004

DESCRIPTION

Jumper N1-1561
Jumper a 52- contact onto the control circuit for 1-FP-C-1. This change does not affect the normal start permissive of the air compressor motor, but it will be locked in until it is turned off by the normal pressure switch.

SAFETY EVALUATION SUMMARY

Troubleshooting of the fire protection hydopneumatic tank control system has not been able to correct a recurrent problem. The level control and pressure control portions of the circuit "fight" against each other. This change will cause the compressor motor to lock in until the correct operating pressure is reached.

SAFETY EVALUATION NUMBER 93-SE-JMP-005

DESCRIPTION

Jumper No. N2-1038
Alarm 2J-F8 ("UNIT 1 SW KEYLOCK SWITCH DEF") will be cleared via patchcord removal and installation of a shorting plug.

SAFETY EVALUATION SUMMARY

The alarm is locked in due to the removal of breaker power from 1-SW-MOV-103A. This jumper will clear the alarm.

SAFETY EVALUATION NUMBER 93-SE-JMP-006

DESCRIPTION

Temporary Modification 1562

SAFETY EVALUATION SUMMARY

One of the P250 computer remote printer cables will be used as a communications link from the control room to a PC in the TSC.","This allows the establishment of an existing communications link without the need to block security/fire doors open while routing temporary cables

SAFETY EVALUATION NUMBER 93-SE-JMP-007

DESCRIPTION

Jumper N2-1040

SAFETY EVALUATION SUMMARY

A camera will be installed in the "C" motor cube to monitor leakage at 1" thermal barrier test connection flange on "C" RCP.
Monitor leakage at 1" thermal barrier test connection flange.

SAFETY EVALUATION NUMBER 93-SE-JMP-008

DESCRIPTION

Jumper N1-1567
The inputs from the main turbine CO2 system into the "CO2 SYSTEM TROUBLE" annunciator will be jumpered out.

SAFETY EVALUATION SUMMARY

Currently the fuses for the turbine CO2 system are removed due to the work being performed on the turbine. This is causing the "CO2 SYSTEM TROUBLE" annunciator to alarm. One of the initial conditions in 0-PT-104.2 is that the annunciator is clear. Therefore, in order to perform the PT the jumper will be installed.

1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

1.0 1.1 1.25 1.4 1.6 1.8 2.0 2.2 2.5 2.8 3.2

150mm

6"

SAFETY EVALUATION NUMBER 93-SE-JMP-009

DESCRIPTION

TEMPORARY MODIFICATION #93-1041
Deletion of the "MANUAL NDT PROT REQUIRED" alarms in the Unit 2 control room, and the deletion of the references to the alarms in the UFSAR section 5.5.8.2 and Figure 5.5-9.

SAFETY EVALUATION SUMMARY

Under the current NDT protection philisophy, which has been reviewed and approved by the NRC, the alarms are not required to meet the North Anna Unit 2 Design Basis requirements.

SAFETY EVALUATION NUMBER 93-SE-JMP-010

DESCRIPTION

Jumper N1-1568
Jumper the input from an installed spare RTD, 1-RC-TE-1432H, into the circuitry for C loop, wide range T hot, replacing 1-RC-TE-1433

SAFETY EVALUATION SUMMARY

The RTD for C loop wide range T hot has failed. This is an Appendix R indication and the installed spare RTD will provide the necessary indications.

SAFETY EVALUATION NUMBER SE-93-JMP-011

DESCRIPTION

Temporary Modification N1-1569

Installation of a stainless steel trough to channel leakage from a dripping conoseal off the Reactor Vessel Head area to prevent accumulation of leakage on or around the Reactor Vessel Flange and bolts.

SAFETY EVALUATION SUMMARY

Prevent accumulation of leakage on or around the Reactor Vessel Flange and bolts.

SAFETY EVALUATION NUMBER 93-SE-JMP-012

DESCRIPTION

Jumper number: 1570

The UFSAR states that a deluge type fire protection system is available in the bearing cooling tower. During the time period that fire protection is isolated, all four fan trips will be jumpered to allow continued operation of the BC system and a fire watch will be posted to provide fire detection and suppression capability.

SAFETY EVALUATION SUMMARY

The performance of the BC system will not be adversely affected. In fact, reliability will be enhanced as unneccessary fan trips will be prevented.

SAFETY EVALUATION NUMBER 93-SE-JMP-013

DESCRIPTION

Temporary Modification N2-1043

The contact in 02-QS-LS-204C will be jumpered out.

SAFETY EVALUATION SUMMARY

02-QS-LS-204A,B and C are level switches in the sumps located in the QS pipe tunnel. When a high water level in any of the sumps is reached, the contact opens and the blue light located in the Aux FW pump house illuminates. The contact on 02-QS-LS-204C is burned up causing the light to remain lit continuously. There are no replacement parts in stock, therefore the contact will be jumpered out. This will return the light to operable (i.e. it will illuminate on a high level as detected by one of the other two switches).

SAFETY EVALUATION NUMBER 93-SE-JMP-014

DESCRIPTION

Jumper TM 93-1571

Defeat the low discharge pressure pump auto-start for the Unit 1 standby Charging pumps.

SAFETY EVALUATION SUMMARY

To allow maintenance on a weld leak near the root valve (1-CH-284) for 1-CH-PI-1121, Charging Header pressure transmitter, per WO 161893.

SAFETY EVALUATION NUMBER 93-SE-JMP-015

DESCRIPTION

Jumper 93-N1-1573

This activity allows the gagging of relief valve 1-SD-RV-102C by providing an alternative relief flow path.

SAFETY EVALUATION SUMMARY

The purpose of gagging via injection of leak sealant into the pilot valve is to stop external leakage. Flow through an alternative path is established to provide piping protection in lieu of the relief valve.

SAFETY EVALUATION NUMBER 93-SE-JMP-016

DESCRIPTION

Temporary Modification N2-1046

Installation of a portable AC unit in the Unit 2 alleyway (located outside of the Unit 2 Rod Drive Room with adequate clearance from the Hydrogen Recombiner block wall). Installation of a ventilation trunk from the portable AC unit to the doorway of the Unit 2 Rod Drive Room. This temporary arrangement will be used to provide additional cooling air to the Unit 2 Rod Drive Room, Cable Vault, and Cable Tunnel areas.

SAFETY EVALUATION SUMMARY

Unit 2 Rod Drive Room AC unit, 2-HV-AC-163, has one of its two compressors out of service due to a shorted winding. The AC unit is currently operating with only one compressor. Due to hot weather conditions, it is desired to provide additional cooling to the affected areas until 2-HV-AC-163 is repaired and returned to a fully operable status.

SAFETY EVALUATION NUMBER 93-SE-JMP-017

DESCRIPTION

JUMPER N93-1575

To supply makeup water to the control room chillers if required while maintenance is being performed on the system.

SAFETY EVALUATION SUMMARY

In order to replace the existing PCV (01-HV-PCV-1303) with a manual valve the expansion tanks (01-HV-TK-6A/B) in the CD subsystem for the control room chillers will be isolated from makeup sourece. In addition the "A" expansion tank will be isolated from the CD subsystem. In order to provide makeup capability for the CD system during this period, a red rubber hose will be installed from a DW connection at the 307 switch gear room (01-DW-142) to the suction of the B and C chiller s (via 01-CD-185 and 206) if makeup is required.

SAFETY EVALUATION NUMBER 93-SE-JMP-018

DESCRIPTION

Temporary Modification N1-1576

The "C" charging pump is being overhauled. In order to improve the environmental conditions for the craft a portable AC unit will be installed.

SAFETY EVALUATION SUMMARY

Installation of a portable AC unit to provide cooling air to the unit 1 "C" charging pump cubicle while maintenance is being performed on the pump. The exhaust duct for the cubicle will be blocked to prevent the conditioned air from being removed.

SAFETY EVALUATION NUMBER 93-SE-JMP-019

DESCRIPTION

Temporary Modification N1-1577

This modification allows for rerouting IA tubing and duct work for the new instrument shop.

SAFETY EVALUATION SUMMARY

The linkage from 1-HV-AOD-184 to the cold deck damper on 1-HV-AC-4 will be disconnected. This will allow the cold deck and hot deck dampers to remain in the correct position for comfortable supply air to the Control and Relay rooms during work on the IA tubing which controls the damper.

SAFETY EVALUATION NUMBER 93-SE-JMP-020

DESCRIPTION

Jumper N2-1050

Jumper power around relay 74-2ENSJ08 in cabinet 2-EP-CB-28BX

SAFETY EVALUATION SUMMARY

Relay 74-ENSJ08 is burned out. The function of this relay is to provide indication of power to the SI/CDA load shed circuitry. During the removal of this relay, continuity of the circuitry will be lost to other components in the cabinet unless the jumper is installed.

SAFETY EVALUATION NUMBER 93-SE-JMP-021

DESCRIPTION

Temporary Modification N1-1578

Maintenance is being performed on the "B" charging pump. In order to improve the environmental conditions for the craft a portable AC unit will be installed.

SAFETY EVALUATION SUMMARY

Installation of a portable AC unit to provide cooling air to the unit 1 "B" charging pump cubicle while maintenance is being performed on the pump. The exhaust duct for the cubicle will be blocked to prevent the conditioned air from being removed. The exhaust duct cover will also maintain flows on operable HHSI/CH pumps.

SAFETY EVALUATION NUMBER 93-SE-JMP-022

DESCRIPTION

Jumper N2-1051

Power fluctuations on the ERFCS UPS power distribution (especially on the C phase) may be the result of power fluctuations on the 2G2 bus.

SAFETY EVALUATION SUMMARY

Install a line disturbance analyzer in cabinet 2-EP-BKR-25G4 (SPARE) on device LD locations 2,4 and 6. Provide FUSED leads to line disturbance analyzer from device LD.

SAFETY EVALUATION NUMBER 93-SE-JMP-023

DESCRIPTION

Jumper N1-1580

Jumper the input from an installed spare RTD, 1-RC-TE-1412F, into the circuity for A loop wide range T cold, replacing 1-RC-TE-1410.

SAFETY EVALUATION SUMMARY

The RTD for A loop wide range T cold has drifted low. The installed spare RTD will provide the necessary indications.

SAFETY EVALUATION NUMBER 93-SE-JMP-024

DESCRIPTION

TM-1052

A brace will be installed on the broken compensator spring housing of 2-RC-MOV-2593.

SAFETY EVALUATION SUMMARY

2-RC-MOV-2593 must be stroked closed to support the outage schedule. Parts are not available at this time to support repairs. The brace will allow the MOV to be stroked closed. Repairs will be performed at a later date.

SAFETY EVALUATION NUMBER 93-SE-JMP-025

DESCRIPTION

Jumper N2-1053

Jumper out CW pump vacuum priming level switches at the CWP breaker cube.

SAFETY EVALUATION SUMMARY

This jumper will allow starting a CW pump after locally verifying vacuum priming exists in the event the CW-VP logic is not made due to a faulty switch.

SAFETY EVALUATION NUMBER 93-SE-JMP-026

DESCRIPTION

Jumper 93-N2-1054

This activity allows the gagging of relief valve 2-SD-RV-202C by providing an alternative relief flow path.

SAFETY EVALUATION SUMMARY

The purpose of gagging via injection of leak sealant into the pilot valve is to stop external leakage. Flow through an alternative path is established to provide piping protection in lieu of the relief valve.

1993 50.59 SAFETY EVALUATIONS REPORTABLE TO THE NRC

JUSTIFICATION FOR CONTINUED OPERATION (JCO's)

93-SE-JCO-001
93-SE-JCO-002
93-SE-JCO-003
93-SE-JCO-004
93-SE-JCO-005

SAFETY EVALUATION NUMBER 93-SE-JCO-001

DESCRIPTION

This SE evaluates the concern of Post-LOCA backleakage via the charging pump suction from VCT check valve 1-CH-215 (2-CH-153).

SAFETY EVALUATION SUMMARY

Leakage can be detected by the operator from both VCT level and pressure alarms once in-leakage to the VCT begins. The leakage could be stopped by remote action using charging system MOVs and still maintain safety injection operability. The leakage could be stopped by local action using the seal water heat exchanger manual isolation. The Unit 1 check valve was recently tested and found to leak 0.06 gpm. The Operations Department was briefed on this event.

SAFETY EVALUATION NUMBER 93-SE-JCO-002

DESCRIPTION

The mainifolds for flow transmitter 1-RC-FC-1481A and 1-RC-FC-1482A were changed during the steam generator replacement outage with manifolds supplied with female threaded tube connections which do not comply with NAI-0001 requiring all welded connections except to the transmitter.

SAFETY EVALUATION SUMMARY

The mainfold has been successfully pressure tested to 2300 psig with no leakage. The compression fittings will be checked with a "go/no go" gauge to ensure proper connection. The manifold connections will be verified during the unit startup containment walkdown. The pressure rating of the fittings is 4700 psig. The manifold and valves are seismically rated to withstand 6G. A failed fitting connection would yield a 3/8" break which would not compromise pressurizer level. The manifold will be replaced during the next refueling outage with a manifold using welded connections.

SAFETY EVALUATION NUMBER 93-SE-JCO-003

DESCRIPTION

Verify that the structural condition of the Unit 1 equipment hatch platform as modified for the steam generator replacement project was adequate to support the hatch missile barrier.

SAFETY EVALUATION SUMMARY

Analysis of the modified equipment hatch platform discovered that portions of the structure are stressed beyond design allowable. Analysis concluded that the platform will not collapse and that the missile barrier will continue to provide protection.

SAFETY EVALUATION NUMBER 93-SE-JCO-004

DESCRIPTION

Investigate the implications of the Salem Rod Control failure of 5/27/93 on North Anna.

SAFETY EVALUATION SUMMARY

Power operation could continue with a postulated failure that could result in a Condition III event, Single Rod Withdrawal At Power. This is contrary to the UFSAR which states that no single failure could cause the accidental withdrawal of a single rod control cluster at full power. A Single Rod Withdrawal at Power (SRWP) will yield reactivity and power peaking responses which are bounded by previous analysis results. A review of all applicable Chapter 15 events shows that the SRWP is the most severe, bounding all other rod misposition events (including misaligned and dropped rods, or a bank withdrawal). Further, even with a single failure as an accident initiator, the SRWP remains within the frequency range of Chapter III events, so that its previous analysis acceptance criteria remain applicable. Existing Tech Specs remain fully adequate to maintain and verify rod operability; misalignment is limited to +/-12 steps and the rod deviation monitor is required to be OPERABLE in Modes 1 and 2. Finally, the ability of the control rods to insert fully within their TS time limit, when demanded by a Reactor Trip signal, remains unaffected.

SAFETY EVALUATION NUMBER 93-SE-JCO-005

DESCRIPTION

JCO 93-005 was written to document that the Unit 2 RCP underfrequency reactor trip was operable based on DR N93-1195 and the actions associated with it.

SAFETY EVALUATION SUMMARY

The neutral designation and routing of the DC power supply to the underfrequency aux relay panels was not consistent with the design basis in the UFSAR. One channel was declared inoperable to eliminate the potential for a single failure resulting in the loss of two channels.