SPECIAL TESTS

NORTH ANNA POWER STATION UNIT NO. 1

COMPLETED IN 1979

1-ST-6: 1H/1J Diesel Generator Load Rejection Test

Description: The purpose of this test is to demonstrate that each emergency diesel generator is capable of rejecting a load of $\geq$ 610 kw without tripping and to verify that the auto-connected loads do not exceed 3000 kw.

Each generator is brought up to speed and loaded to $\geq$ 750 kw. This load is then removed and the peak RPM recorded. The auto-connected loads after a SI and CDA are verified to be $\leq$ 3000 kw by referring to the generator loading schedules in the Final Safety Analysis Report.

SUMMARY OF SAFETY ANALYSIS: This special test will not increase the probability of the occurrence of an accident nor will it increase the probability of a safety related equipment malfunction.

1-ST-7: Recirculation Spray System Pump Special Test

Description: The purpose of this test is to obtain data to verify dependability of pump motor bearings. A dike was installed around the recirculation spray sump and filled with water and the discharges of the pumps were directed back into the sump. The pumps were run for 12 hours each while the bearing and motor casing temperatures were being monitored.

SUMMARY OF SAFETY ANALYSIS: The probability of the occurrence of an accident or malfunction of this safety-related equipment will not be increased by this test. Also the possibility for an accident or malfunction of a different type other than any evaluated previously in the safety analysis report has not been created nor is the margin of safety, as defined in the basis for any technical specification, reduced.

1-ST-12: Recirculation Spray System Pump Special Test

Description: The purpose of this test is to obtain data to verify dependability of pump motor bearings following bearing modification. A dike was installed around the recirculation spray sump and filled with water and the discharge of the pumps was directed back into the sump. The pumps were run for at least 24 hours with the lower bearing temperatures stable for at least 4 hours, while the temperatures were being monitored.

SUMMARY OF SAFETY ANALYSIS: The probability of the occurrence of an accident or malfunction of this safety-related equipment will not be increased by this test. Also the possibility for an accident or malfunction of a different type other than any evaluated previously in the safety analysis report has not been created nor is the margin of safety, as defined in the basis for any technical specification, reduced.

LIST OF DESIGN CHANGES

COMPLETED IN 1979

DC #	Date
78-03	12-18
78-14	10-15
78-15	12-18
78-21	12-31
78-28	12-12
78-44	12-22
78-60	1-5
78-69	12-18
78-70	12-28
78-76	1-2
78-84	4-18
78-85	3-5
79-01	3-28
79-S05	4-16
79-06	12-31
79-S07	12-13
79-S09	12-14
79-S10	12-12
79-S13	4-19
79-S14	12-18
79-17	4-18
79-S18	4-13
79-S19	4-16
79-S21	11-7
79-S22	5-7
79-S23	12-27
79-24	12-12
79-S26	4-29
79-S27	4-18
79-S28	12-21
79-S29	4-28
79-32	12-18
79-S37	12-20
79-38	12-18
79-S48	12-28
79-S50	12-19
79-S51	12-31
79-S53	12-19
79-S56	12-21
79-S57	12-27
79-S58	12-27
79-62	12-27

FACILITY DESIGN CHANGES

NORTH ANNA POWER STATION UNIT 1

COMPLETED IN 1979

1. DC 78-03 Tilting Disc Check Valve Modification (CC)

Description: Three tilting disc check valves supplied by Crane Manufacturing Company were reported by the supplier to be susceptible to excessive vibration allowing the disc to become free and unrestrained in the valve body. The subject valves are located on the component cooling supply line to the thermal barriers for each of the reactor coolant pumps (1-CC-111, 146, 181.)

The resolution was to install a new higher strength pivot pin supplied from a crane retrofit kit.

SUMMARY OF SAFETY ANALYSIS: The probability of the occurrence of an accident or malfunction to safety related equipment will not be increased by this design change. This change will greatly increase system reliability. The possibility of increasing the likelihood of an accident is not created because the design change will not degrade system design.

The margin of safety is not reduced by this change because the change does not adversely effect the component cooling system operation as described in F.S.A.R.

2. DC 78-14 CRDM Shroud Cooling Fan Duct Modification

Description: No adequate support was available for securing the CRDM cooling fan motor during removal and installation when periodic maintenance was required on the fan motor.

The resolution was to provide an access hole in the duct above each fan motor. During normal operation cover plates will be in place over the access holes to prevent loss of cooling air and provide structural rigidity.

SUMMARY OF SAFETY ANALYSIS: The change will be incorporated into a non-safety related system with no additional system interfacing, therefore the probability of an unreviewed safety question is neither created nor increased. The margin of safety is not reduced since the system is not related to safety nor does it affect a safety related system.

3. DC 78-15 Part Length Rod Removal Modification

Description: In the past, use of the part length rods for reactivity control has been prohibited by Tech. Spec. 3.1.3.7. The basis was to eliminate the potential for adverse core power shapes which could have resulted if the part length rods were inserted into the core.

The resolution was to remove the part length rods using approved Westinghouse procedures and to install permanent anti-rotational devices on the associated drive motors. The group step counter indication as well as the individual rod position indication were removed from the bench board in the control room. Tech. Spec. 3.1.3.7 as well as any reference to part length rods was deleted from the Unit 1 technical specifications.

This design change has eliminated all maintenance and refueling concerns which would otherwise be required in conjunction with the part length rods.

SUMMARY OF SAFETY ANALYSIS: According to a letter received from the Westinghouse project engineer, no additional safety analysis is required for removal of the part length rods. The equipment to be used for this modification shall be purchased from Westinghouse and installed by an approved procedure. Therefore, the "unreviewed safety question" posed in 10 CFR 50.59 has been answered by the NSSS vendor, Westinghouse, from whom the modification package shall be purchased.

4. DC 78-21 "B" Stripper Vent Chiller Modifications

Description: A problem with water carryover has been experienced with the gaseous waste system. Inadequate drainage of the stripper vent chiller (1-BR-E-9A, B) was believed to have been the major contributor to this problem. In addition, the vents from the primary drain transfer tank were piped to the outlet of the stripper vent chiller resulting in carryover of water.

The "B" vent chiller (1-BR-E-9B) was modified to add a new 3/4" nozzle on the top of the chiller to serve as the gas outlet to the stripper compressors. Gas and steam will enter the top of the chiller through the existing 2" nozzle with the 3/4" nozzle on the bottom of the chiller to be used as a drain to the suction of the circulation/discharge pumps (1-BR-P-7B & 10B). Also, in the future, operating personnel should make sure that the vents from the VCT and PDTT are passed through the vent chiller in order to remove water before reaching the stripper gas compressors.

Only the "B" vent chiller was modified at this time since this portion of the system had not been operated and was not expected to be contaminated.

SUMMARY OF SAFETY ANALYSIS: The stripper vent chiller modifications do not create an "unreviewed safety question", as defined in 10 CFR 50.59.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, previously evaluated in the safety analysis report, is not increased because the modified piping is erected to the original code requirements. The modificiations should result in improved system performance.

5. DC 78-28 Outside Recirculation Spray Pump Metal Expansion Joints

Description: Discharge flange alignment is quite critical for the ORS

pumps. Any misalignment in this flange could cause excessive loads on the pump discharge nozzles.

The resolution was the installation of metal expansion joints and the modification of pipe restraints and hangers.

SUMMARY OF SAFETY ANALYSIS: Installation of metal expansions at the recirculation spray pumps located outside the containment does not create an "unreviewed safety question" as defined in 10 CFR 50.59. The original design basis of the recirculation spray system has not been changed.

6. DC 78-44 Reactor Coolant System Overpressurization Protection Modification:

Description: The intent of this change was to provide a redundant control system to actuate the existing pressurizer power-operated relief valves (PORV) to protect the reactor coolant system (RCS) from overpressurization transients during low temperature operation when the vessel is susceptible to brittle fracture.

A new pressure transmitter (PT-402-1) was added to the reactor coolant system in the same instrument tubing connection feeding PT-402, and is used to control PCV-1455C. Additional pressure switches were added to existing PT-403 to be used in controlling PCV-1456. Temperature switches were added to TE-410, 420, and 430 (cold leg RTD's on each loop). At the proper pressure set point, as determined by a low auctioneered temperature signal generated from the new temperature switches, a trip signal is generated for the PORV's. This open signal will energize a new solenoid operated valve (SOV-3) which isolates containment pressure reducing regulators to the PORV in order to open it. The nitrogen system taps off an existing high pressure nitrogen supply line and will have two nitrogen accumulators each sized to cycle its associated PORV 120 times upon a loss of nitrogen supply header pressure. If the nitrogen supply pressure drops below 1950 psig in either accumulator tank, the control room operator is alerted by annunciators (windows 1C-53 and 1C-62).

Two three position key lock switches (auto-off-open), the operation of which will be administratively controlled, were added to the benchboard in the control room. During normal operation, the key switches will be in the "off" position, and the PORV's will open at their normal setpoint (2335 psig). When the RCS cold leg temperature is less than 320° F and the keyswitch is in "off" or the in-line MOV isolation valve is not fully open, an annunciator (window 1C-29) will tell the control room operator that solid water NDT protection is required. At the same time, two other annunciators (windows 1C-37 and 1C-145) will alarm telling the operator to place the key switches for the PORV's to the "auto" position provided the pressure is low enough not to cause the relief valves to open. Therefore, all three annunciators should be lit before the "auto" position is selected on the keyswitches. If the latter two annunciators are not lit, RCS pressure must be lowered below the PORV lift setpoints (see table below) before arming the solid water overpressure protection system.

Once the keyswitches are placed in the "auto" position, the PORV's will open if the RCS pressure reaches the setpoints given below. At the same time, the control room operator will be alerted by an annunciator (window 1C-30) telling him that the system had been actuated. The setpoints for Unit 2 are different than those for Unit 1 and are shown in parenthesis.

With the keyswitches in Auto:

	Tc* 140°F	Tc* 140°F
PCV 455C	425 psig (400 psig)	500 psig (470 psig)
PCV 456	410 psig (385 psig)	485 psig (455 psig)

*TC is auctioneered low cold leg temperature

The selection of the temperature at which low temperature overpressure protection is required was based on the material properties of the reactor coolant system as exemplified by Figures 3.4-2 and 3.4-3 in the Technical Specifications. The temperature was chosen by determining the point (320° for Unit 1) below which the pressurizer safety valves afforded no NDT protection to the RCS. Over core life, as the reactor vessel is exposed to neutron flux fields, the material properties of the vessel will change. The above mentioned curves will move to the right requiring a change in the temperature below which low temperature overpressure protection is needed. This means that the temperature at which this protection will be required will have to be reevaluated periodically, resulting in changing certain instrument setpoints.

The material properties for Unit 2 reactor vessel are different than those for Unit 1. In fact, the pressure-temperature heatup and cool down curves are more limiting for Unit 2. Between 320° and 340°, the safety valves do not afford sufficient protection. As a result, changes had to be made to the overpressure protection scheme for Unit 2. Two redundant annunciators (windows 2D-43 and 2D-44) were installed to tell the control room operator when manual initiation of NDT protection was required (i.e. manually opening a PORV, shutting off a charging pump, etc.). The alarms are actuated when RCS temperature is below 340° and RCS pressure is above 500 psig. Other than these two alarms, the operation of Unit 2 low temperature overpressure protection system is identical to that for Unit 1.

SUMMARY OF SAFETY ANALYSIS: The reactor coolant system overpressure protection modification does not create an "Unreviewed Safety Question" as defined by 10 CFR 50.59. The modification ensures that under solid water mode 4 and 5 operations there is automatic overpressure protection. The new portions of the system which are Class 1E cannot be degraded by the existing controls which are not safety related. All added mechanical equipment, including the nitrogen accumulator, piping, and valves is designed to seismic Category I criteria and is designed to prevent generation of internal missiles. The system is designed to meet the single failure criterion during both the solid water and normal mode of operation. Further, a detailed design description and safety analysis have been submitted to the NRC in response to NRC Staff Comment 5.81, and by letter Serial No. 214/032179 dated April 17, 1979.

7. DC 78-60 MoistureSeparatorReheater High Level Turbine Trip and MOV-AS-100 Actuation

Description: Westinghouse does not recommend that the turbine be tripped from high level in the MSR if there are no other indications of distress. The reasons for this are discussed in the Westinghouse letter included in the Engineering Review.

SUMMARY OF SAFETY ANALYSIS: The removal of the turbine trip and MOV-AS-100 actuation does not create an "unreviewed safety question" as defined in 10 CFR 50.59. This system is non-safety related.

8. DC 78-69 Emergency Bus Degraded Voltage Modification

Description: In the past, the undervoltage protection for the 4160V emergency buses consisted of one level of protection designed for loss of voltage detection. Incidents at other plants had demonstrated that it was possible for a degraded (reduced) voltage condition to exist for periods long enough to cause damage to safety-related equipment.

To remedy this situation, not only was a degraded voltage system added to detect sustained undervoltage conditions, but modifications were also made to the existing undervoltage system. Three potential transformers (connected phase to ground) were installed on each emergency bus. This replaced the two potential transformers (connected phase-to-phase) which had formerly been used on each emergency bus. The potential transformers feed three sets of three voltage relays.

One set of voltage relays is used to detect degraded voltage and therefore, is set to operate at 90% nominal voltage. When 2/3 of the voltage relays detect degraded voltage, a timer is energized which times for 60 seconds and then starts the diesel, isolates the emergency from the reserve station service, and sheds all loads from the emergency bus except for the charging pumps, the low head safety injection pumps, and the feed to the 480V loads. After the diesel comes up to speed and its output voltage reaches 95% of nominal, the diesel output breaker closes and the loads are sequentially connected to the emergency bus. If a safety injection signal is present during the degraded voltage condition, another timer is energized which times for 9 seconds before sending the diesel a start signal (which it should have already received from the SI) and isolating the emergency bus.

The other two sets of voltage relays are used for the undervoltage protection scheme. One of these sets of undervoltage relays is set to operate at $<$ 74% nominal voltage such that when 2/3 of the relays detect undervoltage, a timer is energized and times for 1.2 seconds before starting the diesel. The last set of voltage relays are set to operate at $<$ 72% nominal voltage such that when 2/3 of the relays detect undervoltage, a timer is energized and times for 2.0 seconds before isolating the emergency bus and shedding all loads.

System at the North Anna Power Station were retrofitted with new Lead Lag Cards, supplied by Westinghouse, to prevent perturbation of the cards output. This design change was for the installation of new Lead Lag Cards into the Process Instrumentation and Control System. These new Lead Lag Cards prevent the possibility of perturbation of the cards output.

SUMMARY OF SAFETY ANALYSIS: Since the total affect on the Process Instrumentation and Control System is to prevent perturbation of the cards output by replacing a zener diode and a resistor, the probability of an accident occurring is not increased. This change will prevent the agitation of the lead lag cards internal circuitry but will allow the normal operation of the lead lag card to occur.

The agitation is caused by a step change in the power supply voltage which the replaced zener diode and resistor will eliminate. The addition of the zener diode and resistor to the Instrumentation and Control System will enhance the reliability of the instrumentation circuits.

12. DC 78-85 Charging Pump Lube Oil Cooler Flow Transmitter Modification

Description: The flow transmitters and indicators, FT-SW-101, A, B, C and FI-SW-101, A, B, C were removed and reworked so that the range of the Charging Pump Lube Oil Cooler Flow Transmitters is 0-60 gpm instead of the 0-40 gpm range that it used to be. In the transmitters, FT-SW-101, A, B, C, the 0-205" H_2O range capsules for the 0-40 gpm range were replaced with high range capsules good for 0-850" H_2O, because at a 60 gpm flow, orifice plates FE-SW-101, A, B, C will produce a 265.5" H_2O differential. In the indicators, FI-SW-101, A, B, C, the 0-40 gpm scale cards were replaced with new scale cards graduated 0-60 gpm sqrt. After calibration and installation, the transmitters and indicators were placed back in service.

SUMMARY OF SAFETY ANALYSIS: Replacing the range capsules in FT-SW-101 A, B, C and the scale cards in FI-SW-101, A, B, C will not create an "unreviewed safety question", as defined in 10 CFR 50.59.

13. DC 79-01 Vacuum Priming Pump Auto Start Circuit Modification

Description: The condenser water box vacuum priming pumps were modified so that the automatic start circuit is not powered from the -1A vacuum priming pump. This modification was accomplished by removing one of the wires from the present power supply located in pump breaker, MCC 1B2-2 (1-EP-MC-03), Section C-1, Ckt. 1VPSA01, 1A (wire 1VPSA01X00-B). Wires were added to supply power to the automatic start circuit from 1-EP-CB-28G2 (AR-G2), TBG 10 and TBG 12; to 1-ET-CB-28G1 (AR-G1), TD4 and TD5. Wire was removed connecting AL-M1 to AH1-6 in AR-G1 and wire was added from AH1-6 to TD5. Cable identification tags were replaced as required following this modification.

SUMMARY OF SAFETY ANALYSIS: This change will provide a more reliable power supply for the automatic start circuit of the vacuum priming pumps. The possibility of increasing the likelihood of an accident is not created since these pumps are not safety related.

Once the emergency bus is isolated from the reserve station service and the diesel output breaker closes, the load shedding feature initiated from either a degraded or undervoltage condition is defeated. This is necessary to prevent voltage drops encountered during the diesel loading sequence from causing load shedding and thus negating the loading of the diesels.

A new annunciator (window 1F-40) was added to Unit 1 to alert the control room operator when a degraded voltage condition (90% voltage) exists for greater than 15 seconds or when an overvoltage condition (4400 V with no time delay) exists on either emergency bus. The new voltage relays added for degraded voltage protection also have contacts which actuate on overvoltage, and it was one of those contacts which was used in the contact development for this annunciator. It was also discovered that upon re-energizing the emergency buses after they went dead, that the overvoltage contact will sometimes operate resulting in a spurious alarm which can be cleared by resetting the annunciator. On Unit 2, two separate annunciator windows were used. Window 2F-40 annunciates for degraded voltage while window 2F-63 annunciates for the overvoltage condition.

SUMMARY OF SAFETY ANALYSIS: The degraded voltage modification does not constitute an unreviewed safety question as defined in 10 CFR 50.59. The modification provides an additional level of undervoltage protection to protect electrical equipment during a sustained degraded voltage condition. It does not affect system reliability or capability.

9. DC 78-70 Replacement of Limit Switches on Inside Containment Isolation Valves

Description: The limit switches on 24 containment isolation valves located inside the containment were not qualified for all containment accident conditions. These limit switches feed valve status lights in the control room.

The resolution was to replace the limit switches with switches qualified for all containment conditions.

SUMMARY OF SAFETY ANALYSIS: Replacement of the limit switches on containment isolation valves inside the containment does not create an "unreviewed safety question" as defined in 10 CFR 50.59. The new limit switches will meet the appropriate environmental and seismic qualification requirements. The original switches were not required to meet environmental conditions.

10. DC 78-76 Service Water Guy Wire Replacement

Description: This design change package is for the implementation of the change out of all guy wire and associated hardware used on North Anna Unit 1 service water spray piping.

SUMMARY OF SAFETY ANALYSIS: This activity does not create and "unreviewed safety question" as defined in 10 CFR 50.59. This work will not change the station design and will not render a complete loop of the service water system inoperable.

11. DC 78-84 Retrofit of Lead/Lag Cards

Description: The Lead Lag Cards in the Process Instrumentation and Control

14. DC 79-S05 Service Water/Component Cooling Water Extended Temperature Range Support Work

Description: This design change implements hanger modifications to the North Anna Unit 1 service water and component cooling water piping system. These modifications are needed to accommodate the increased loads on hangers resulting from restress analysis for a widened service water temperature range. The new design temperature corresponds to a service water reservoir temperature range of 30° F to 115° F. Because of the change in service water temperature, component cooling water will also require reanalysis.

SUMMARY OF SAFETY ANALYSIS: The Stress Analysis on large and small bore piping and hangers does not create an "unreviewed safety question" as defined in 10 CFR 50.59. The original design basis of the Service Water System have not been changed. The level of integrity of the Service Water System is not changed.

15. DC 79-06 Reactor Containment Purge System Modification

Description: The reactor containment purge system consists of dual, paralleled supply and exhaust fans arranged to ventilate either containment when the pressure has been raised to within 1" W.G. of atmosphere. The supply and exhaust ducts are fitted with butterfly valves on both sides of the containment penetrations for pressure integrity. A high radiation signal from the Unit 1 containment gas or particulate monitors will automatically trip the containment purge supply and exhaust fans and close the containment ventilation butterfly valves, thus isolating the containment and also preventing the ventilating of Unit 2 containment for refueling or maintenance.

Design Change Request DC 79-06 describes a change to be made to allow use of common supply and exhaust fans for ventilating either containment while the other containment is isolated either at power or with a high radiation signal. To accomplish this, spare contacts from each unit containment purge isolation valve limit switches will be wired in parallel with the respective high radiation signals in the supply and exhaust fans control circuit.

SUMMARY OF SAFETY ANALYSIS: Modification of the control logic for the Containment Purge System does not create an "unreviewed safety question" as defined in 10 CFR 50.59. The modification introduces a parallel control logic into the system which allows independent operation of the Containment Purge System for one unit (during refueling) provided the containment isolation of the second unit from the purge system is maintained. The modification provides independence between the purge functions for Units 1 and 2, but does not affect containment isolation capability or reliability.

The purge system containment isolation valves control circuits and the high radiation signal which closes these valves upon activation are not affected nor changed by this modification. Therefore, the modification does not affect equipment important to safety which was previously reviewed in the FSAR. The modification uses existing spare contacts of components which were electrically interlocked to accomplish this design change.

Therefore, the possibility for a malfunction of a different type occurring is negated.

The modification does not affect system capacity, method of operation (normal or abnormal) nor design basis for any of the postulated accidents. Therefore, this modification does not infringe upon the margin of safety.

16. DC 79-S07 Supplementary Neutron Shielding

Description: The radiation levels inside the containment, as determined by radiation surveys in Unit 1, were greater than the design levels presented in Section 12 of the FSAR at two locations. They were the annulus area between the crane wall and the containment wall on the operating level (Elev. 291) at the crane wall openings and inside the personnel lock.

The resolution involved the addition of supplementary neutron shielding in the upper reactor vessel area and in the crane wall openings. Neutron attenuating shield material was placed in the crane wall openings extending from directly opposite the personnel hatch to the stair well entrance and over the portion of the fuel transfer canal behind the crane wall. The shielding added to the upper reactor vessel area is composed primarily of a six segment cylindrical collar assembly which fits around the vessel and rests on the top of the neutron shield tank. In addition, a saddle assembly consisting of U-shaped blanket type covers were installed on top of each nozzle and dust cover blocks were installed under the nozzles to partially fill the space between the dust cover and the collar base underneath each nozzle. The saddles are comprised of approximately 130 1/4" wide strips of silicon based neutron attenuating material per nozzle. To pervent blockage of the refueling canal 6" drain by these strips a domed strainer was installed over this drain.

SUMMARY OF SAFETY ANALYSIS: The addition of supplementary neutron shielding and associated work does not create an "unreviewed safety question", as defined in 10 CFR 50.59.

With the neutron shielding in place, the fuel assembly impact loads have increased from those reported in Appendix 5A by approximately 10 percent. This change alone would reduce margins previously reported; however, the loads are still less than the allowable values. Recent testing on fuel grid impact strength has resulted in Westinghouse increasing the allowable loads by approximately 25 percent above those in Appendix 5A. These new allowables have been previously reported to the NRC on Diablo Canyon docket (Docket Nos. 50-275 and 50-323). When using the new allowable loads along with the revised impact loads, the revised margin is higher than in Appendix 5A, Section 5A.9. The "better estimate" factor of safety of 1.76 would now be approximately 1.97. In addition, the limiting stress on the reactor vessel internals at the core barrel girth weld has decreased from that reported in Appendix 5A. This is due to the time phasing of the component forces.

In summary, the supplementary neutron shield for Unit 1 restores expected dose rates inside the containment to the Section 12 limits, and does not change the conclusions previously established in Appendix 5A.

17. DC 79-S09 Service Water to Charging Pumps

Description: Thermal performance tests have indicated that for multi-unit operation post accident service water reservoir temperatures could reach 110° F. This is higher than the original design criteria. The as-built cooler piping arrangement has also been shown to have restricted flow due to fouling of skid piping.

The resolution involved replacing the lube oil and gear box coolers on each charging pump with larger capacity coolers. Also, new service water return and supply headers were added to serve the gear box and seal coolers. The required service water flow rate to each charging pump is now 35 gpm consisting of 15 gpm to the lube oil cooler, 10 gpm to the gear box cooler, and 5 gpm to each of the two seal coolers.

These modifications necessitated changes in the method of monitoring flow to the charging pump coolers. Orifice type flow elements were replaced because of their associated high pressure drop. At present, service water to the lube oil coolers is monitored by measuring lube oil temperatures with both indication and high temperature annunciation (window 1C-42) being provided in the control room. The service water flow to the gear box and seal coolers, which is supplied by the new headers, is detected by measuring the pressure drop across the paralleled coolers with local flow indication and low flow annunciation (windows 1B-50, 59, 61) in the control room.

SUMMARY OF SAFETY ANALYSIS: The service water to charging pump modification does not create an "unreviewed safety question", as defined in 10 CFR 50.59.

The modified Service Water piping supplies the cooling water flow required by the Charging Pumps and is designed to minimize restricted areas where fouling may occur.

18. DC 79-S10 Reactor Containment Spare Penetration Pipe Caps

Description: The North Anna FSAR, Section 6.2.4.3 requires that all spare penetrations through the containment be sealed at both ends with a welded cap. A review of Unit 1 pipe penetrations revealed that some penetrations are only capped on the containment side.

The resolution was the installation of caps on the Auxiliary Building side of spare pipe penetrations which were not previously capped. These caps were welded in place and equipped with test plugs.

SUMMARY OF SAFETY ANALYSIS: Since this modification brings the physical plant into conformance with the FSAR, an unreviewed safety question does not exist.

19. DC 79-S13 Service Water System Reinforcing Pad

Description: This design change is for the implementation of the installation of a reinforcing pad at the tee connection for 20" WS-406-151-Q3 to 36" WS-2-151-Q3. This change is being made to ensure the adequacy of design of the service water system piping with a temperature range of the service water reservoir of 30° F to 115° F. This change is being made based on the design requirement for power piping built to the requirements of ANSI-B31. 7-1969 with the 1970 addenda.

SUMMARY OF SAFETY ANALYSIS: The installation of a reinforcing pad at Node #86 does not create an "unreviewed safety question" as defined in 10 CFR 50.59. The original design basis of the Service Water System have not been changed. The level of integrity of the Service Water System is not changed.

20. DC 79-14A, B, C, Diesel Generator Exhaust Modification

Description: The diesel exhaust silencer is not seismically restrained or protected against tornado missiles as required by Section 3.2 of the FSAR. Damage to the silencer could restrict the exhaust flow from an operating diesel resulting in the diesel being shutdown.

The resolution involved the installation of a 24" exhaust bypass line with a butterfly isolation valve. This line was afforded protection by the construction of a reinforced concrete and structural steel tornado missile structure on the roof of the diesel rooms. The valve in the bypass line will be used to isolate the bypass and divert exhaust through the silencer while testing the diesel. After visually verifying the position of this valve from above, the chain can be locked in position from the diesel room. Care should be taken to see that this valve is returned to the full open position upon completion of testing.

SUMMARY OF SAFETY ANALYSIS: Installation of properly designed concrete shielding for diesel exhaust does not create an "unreviewed safety question" as defined in 10 CFR 50.59. The original design basis had not been changed.

21. DC 79-17 Gland Steam Condenser Modification

Description: This modification consists of replacing the discharge valves, 1-VP-24 and 1-VP-25, from fans 1-GS-F-1 and 1-GS-F-2 with spacers and installing one valve in the common suction line to the fans. Local pressure instrumentation, PI-SV135 and PI-MS112P, was relocated because of interference and for easy readability from the new operating station.

SUMMARY OF SAFETY ANALYSIS: Modification of the gland steam condenser does not create an "unreviewed safety question" as defined in 10CFR 50.59. The modification moved the valves for vent vacuum adjustment from the discharge piping to the suction piping of the blowers. This modification does not affect safety related equipment.

22. DC 79-S18 Safety Injection Pipe Support Modification

Description: This Design Change covers the implementation of hanger modifications as described below:

Pipe Support No.	Line No.	MSK No.	Modification
FPH-SI-16-1	6"-SI-16-1502-Q1	103AD	New spring settings
FPH-SI-63-4	2"-SI-63-1502-Q1	103AD	New spring settings
FPH-SI-131-1	6"-SI-131-1502-Q1	103N	Base Plate
SI-SH-207	6"-SI-16-1502-Q1	103AC	New spring setting
FPH-SI-21-3	6"-SI-21-1502-Q1	103AJ	Support steel
SI-SH-210	6"-SI-21-1502-Q1	103AC	(2) new springs to replace existing springs

Work performed on this design change is in accordance with the design requirements of ANSI B31.7-1969, and 1970addendaand required stress analysis.

SUMMARY OF SAFETY ANALYSIS: The safety injection pipe support modification does not create an "unreviewed" safety question, as defined in 10 CFR 50.59.

This modification does not change the level of integrity of the Safety Injection System nor the design basis for pipe supports.

23. DC 79-S19 Fisher Control Valve Limit Switch Modification

Description: Limit switch brackets were installed on the limit switch assemblies for Fisher Control Valves.

SUMMARY OF SAFETY ANALYSIS: The probability of the occurrence of an accident or malfunction of this safety related equipment will not be increased by this modification. This modification will make the Fisher Control Valves more reliable.

24. DC 79-S21 Fire Suppression Water Supply to Records Storage Building

Description: The underground fire main loop was modified by the addition of a 4 inch branch supply to the Records Storage Building. This was accomplished by installing a tee in the underground main for connection of the branch line to the Records Storage Building. The branch line is provided with a post-indicating isolation valve.

SUMMARY OF SAFETY ANALYSIS: This modification does not create an "unreviewed safety question" as defined in 10 CFR 50.59.

The yard hydrant piping system is categorized Seismic Class 1, to be available as backup sources for auxiliary feedwater and fuel pool makeup. The design of this modification to the yard loop is consistent with the remainder of the underground main.

25. DC 79-S22 Service Water/Component Cooling Water Extended Temperature Range Support Work

Description: Modification of hangers on four pipes reduces the stress levels on the pipes to an acceptable level. This design change was made based on the design requirements for power piping built to the requirements of ANSI B31.7-1969 with the 1970 addenda.

SUMMARY OF SAFETY ANALYSIS: The stress analysis and resupporting of large and small bore piping and hangers does not create an "unreviewed safety question" as defined in 10 CFR 50.59. The original design basis of the Service Water System have not been changed. The level of integrity of the Service Water System is not changed.

26. DC 79-S23 Service Water/Component Cooling Water Extended Temperature Range Support Work

Description: The original design basis for the service water system was for a maximum service water temperature of 95°F. Recent analysis of accident conditions during multi-unit operation indicates that the service water reservoir temperature would exceed 95°F. This had raised concern about the structural adequacy of the large bore piping and supports for both the service water and component cooling systems.

To ensure adequate design of piping and hangers for these systems, a stress analysis for all large bore piping and hangers was rerun using a service water reservoir temperature range of 30-115°F. After the stress analysis was completed, any piping or hanger which was found to be inadequate for the new temperature range was modified.

SUMMARY OF SAFETY ANALYSIS: The stress analysis and resupporting of large and small bore piping and hangers do not create an "unreviewed safety question", as defined in 10 CFR 50.59. The original design basis of the service water and component cooling water systems have not been changed. The level of integrity of the service water and component cooling water system is not changed.

27. DC 79-24 Turbine Driven Aux. Feedwater Throttle Valve Position Annunciation

Description: The Auxiliary Feedwater Pump (1-FW-P-2) turbine throttle trip valve can be locally tripped by manually actuating the overspeed trip lever. This presents a condition where the trip valve could be shut, disabling the feed pump, without control room indication of the condition.

The resolution involved the installation of a position switch on the trip valve to provide annunciation in the control room whenever the trip valve is not full open. Two annunciator points (windows 1E-55 and 1E-63) were changed to indicate when the trip valve was not open.

SUMMARY OF SAFETY ANALYSIS: The probability of the occurrence of an accident or malfunction to safety related equipment will not be increased by this design change since this change provides the control room with the turbine trip status.

The possibility of increasing the likelihood of an accident is not created since the annunciator is not safety related.

The margin of safety is not reduced by this change since this change does not affect safety related systems.

28. DC 79-S26 Regenerative Heat Exchanger Support Modification

Description: Due to the various loading in the regenerative heat exchanger, the supports were modified to prevent overstressed conditions.

SUMMARY OF SAFETY ANALYSIS: The modification of the regenerative heat exchanger support does not create an "unreviewed safety question", as defined in 10 CFR 50.59. The original design basis of the regenerative heat exchangers have not been changed. The level of integrity of the heat exchanger is not reduced.

29. DC 79-S27 Replacement of MOV-RS100B

Description: MOV-RS100B failed to pass "type C" containment isolation testing. Leakage through the valve was in excess of that allowable and could not be corrected.

The valve was replaced with a 6"-300 lb. gate valve. This use of a 6 inch valve will not alter the original flow characteristic any significant amount as the system line downstream is 4 inches in size and upstream is 6 inches in size.

SUMMARY OF SAFETY ANALYSIS: The replacement of the motor operated valve does not create an "unreviewed safety question." The seismic supports are adequate to support the new valve. The motor starters can handle the new valve and the requirements of Reg. Guide 1.47 have been met.

30. DC 79-S28 Steam Generator Support Temperature Sensor Modification

Description: The terminations for the steam generator support temperature sensors were located within the support blankets. If a problem developed with the temperature sensors during power operation, the support blankets would have to be removed to provide access to the terminations. However, the unit would have to be shutdown and cooled down before the blankets could be safely removed, and this would require three days of cool down time.

In order to prevent having to enter a cold shutdown condition in order to gain access to the sensor terminations, flexible cables were routed from the temperature elements to junction boxes located outside the blankets.

SUMMARY OF SAFETY ANALYSIS: Since the total affect on the Steam Generator Support Temperature Sensors is to relocate the terminations outside the blanket, the probability of an accident occurring is not increased.

By relocating the terminations outside of the blanket, it will make access to and repair of the temperature sensors more rapid and therefore enable more sources of temperatures of the Steam Generator Supports. Since this addition enhances the temperature elements circuits, no possibility of the addition causing an accident exists.

The margin of safety is not reduced due to the fact that the overall reliability of the Steam Generator Support Temperature Sensors is increased.

31. DC 79-S29 Safety Injection Support Modification (MSVH)

Description: This design change is for the implementation of hanger modifications to the North Anna Unit 1 safety injection piping system. These modifications are required due to updated Main Steam Valve House building displacement criteria.

SUMMARY OF SAFETY ANALYSIS: The modification to the Safety Injection piping supports does not create an "unreviewed safety question", as defined in 10 CFR 50.59. The original design basis of the pipe supports has not been changed. The level of integrity of the pipe supports is not reduced.

32. DC 79-S32 Reserve Station Service Power System Modification

Description: As a result of a review of the electrical loading requirements for combined Unit 1 and 2 operation, it was discovered that the three reserve station service (RSS) transformers, the transformer secondary leads, the transfer bus feeder breakers, and the tranfer buses were not capable of continuously carrying full load currents under certain conditions. The conditions of concern involved one unit starting up with the other unit tripping (with and without a CDA) and the simultaneous tripping of both units.

The resolution was to remove all station service bus loads from the transfer buses. This was accomplished by removing the section of bus which tied the normal station service buses to the transfer buses. The normal station service buses were fed from the RSS transformers by newly installed transformer secondary leads. Tubular bus was routed from the secondaries of the RSS transformers to the Turbine Building. From this point, cables were run over the outside of the Turbine Building to the Unit 1 and 2 normal station service buses.

Overload relays and undervoltage relays were added to monitor the secondary of the RSS transformers. The overload relays were set to activate at 4700 amps (113% full load rating). An overload condition lasting for one minute is annunciated in the control room (annunciator windows 1J-32 and 2H-61), and the transformer fans are given a signal to start. Loss of control power to the overload alarm circuit is also annunciated by the annunciators referenced above as is the loss of control power to the load shedding circuit or the fact that the load shedding control switch is in "Defeat."

Two undervoltage relays per RSS transformer are used to monitor the secondary voltage. If the voltage as sensed by both relays drops to 2429 volts (58% of nominal) the corresponding transfer bus feeder breaker and

Unit 1 and 2 normal station service bus feeder breakers are tripped. This condition is annunciated in the control room on both Unit 1 and 2 (windows 1H-23 and 2H-23).

SUMMARY OF SAFETY ANALYSIS: The RSS power system modification does not constitute an "unreviewed safety question" as defined in 10 CFR 50.59. The modification supplies RSS power to the emergency busses via the existing cable feeders and transfer buses and to the normal station buses via separate combination bus and cable feeders and removes these normal station buses from the transfer buses. The modification increases the reliability and capacity of the system.

The modification removes the nonsafety related normal station buses from the transfer buses and feeds them separately; however, the emergency buses will continue to be fed from the transfer buses using the existing installation. Therefore, the modification does not adversely affect the equipment important to safety which was previously reviewed in the FSAR.

All new components and materials used for modification of nonsafety related equipment are totally compatible (relative to design, quality, and functional requirements) with original components and materials. Therefore, the possibility for a malfunction of a different type than previously evaluated does not exist.

The modification will not reduce the capacity, method of operation, or design basis for any safety related systems or component for any of the postulated accidents. Therefore, this modification does not reduce the margin of safety as defined in any technical specification.

33. <u>DC 79-S37 Modification of the Engineered Safety Features Actuation from Pressurizer Level</u>

Description: As a result of the Three Mile Island incident, the NRC requested that automatic initiation of safety injection on low pressurizer level coincident with low pressurizer pressure be changed to delete the low pressurizer level signal. This design change is in response to NRC IE Bulletin 79-06A (Revision No. 1).

The low pressurizer level channels were removed from the Engineered Safety Features protection scheme. The three separate low-low pressurizer pressure channels were connected to a single output card (A417) to provide a two out of three initiation of safety injection.

This change required that several annunciator windows be changed in the control room. All the trip status annunciators relating to pressurizer low level (windows 1L-35, 1L-43, 1L-51, and 1L-37) were deleted. The wording on other trip status annunciators (1L-38, 1L-46, and 1L-54) was changed to indicate the status of the pressurizer low-low pressure channels. Finally, the wording on the first out annunciator (1D-18) for reactor trip caused by low pressure SI was changed.

SUMMARY OF SAFETY ANALYSIS: Since the total affect to the Engineered Safety Features (ESF) protection circuits is to prevent a recurrence of an incident similar to the Three Mile Island incident, the probability of an accident occurring is not increased.

The possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR will not be created. The removal of the low pressurizer water level from the Pressurizer Pressure-Pressurizer Level coincidents for Safety Injection actuation enhances the operation of the ESF protection system. In the past all three pressurizer level channels were in the tripped condition which would result in a Safety Injection if any one pressurizer pressure channels were to receive a low-low pressure indication. This design change changes the present one out of three low-low pressurizer pressure to a two out of three low-low pressurizer pressure actuating a Safety Injection Signal.

The margin of safety is not reduced since the overall reliability of the system is increased.

34. DC 79-S38 Reserve Station Service Load Shedding

Description: Design Change 79-S32 made modifications to the reserve station service (RSS) in order to ensure that no equipment would be overloaded for various two unit operating configurations. However, the impedance of the RSS transformers is high enough to cause low voltage profiles on station emergency and normal buses during certain operating conditions when both units are being fed from the RSS.

The resolution involved the addition of an automatic load shedding scheme for certain non-safety related secondary plant equipment. Load shedding trip signals will be sent to specific pieces of equipment based on the position of a unit startup switch and the status of certain plant equipment. The tripping signal will trip the specified equipment if it is running and block the starting of any specified equipment capable of auto starting. Certain equipment will receive a trip signal any time load shedding is initiated. This equipment is both units' high pressure drain pumps, both units' low pressure heater drain pumps, the tie breaker (15G10) between Unit 1 and 2 G buses, and the 34.5 KV reactor banks in the switchyard. The load shedding sequence for this equipment is initiated if any reserve station service transformer is feeding the normal station service buses on both units at the same time and a feed pump is running for the unit which is selected on the unit startup switch. The unit startup switch has two positions: "Unit 1 Startup" and "Unit 2 Startup".

Other specified equipment (Unit 1 and 2 condensate and main feedwater pumps) will receive trip and lockout signals upon load shed initiation based upon the position of the unit startup switch and which main feedwater pumps are running as delineated below (see Unit 2 FE-21W-1):

CONDITION / UNIT START-UP SWITCH POSITION	15A1 and 25A1 Closed		15B1 and 25B1 Closed		15C1 and 25C1 Closed	
	1-FW-P-1A Running	2-FW-P-1A Running	1-FW-P-1B Running	2-FW-P-1B Running	1-FW-P-1C Running	2-FW-P-1C Running
"Unit 1 Startup"	1-FW-P-1B 1-FW-P-1C 1-CN-P-1B 1-CN-P-1C 2-FW-P-1A 2-FW-P-1B 2-CN-P-1A 2-CN-P-1B	No Load Shedding	1-FW-P-1A 1-FW-P-1C 1-CN-P-1A 1-CN-P-1C 2-FW-P-1A 2-FW-P-1B 2-CN-P-1A 2-CN-P-1B	No Load Shedding	1-FW-P-1A 1-FW-P-1B 1-CN-P-1A 1-CN-P-1B 2-FW-P-1B 2-FW-P-1C 2-CN-P-1B 2-CN-P-1C	No Load Shedding
"Unit 2 Startup"	No Load Shedding	1-FW-P-1A 1-FW-P-1B 1-CN-P-1A 1-CN-P-1B 2-FW-P-1B 2-FW-P-1C 2-CN-P-1B 2-CN-P-1C	No Load Shedding	1-FW-P-1A 1-FW-P-1B 1-CN-P-1A 1-CN-P-1B 2-FW-P-1A 2-FW-P-1C 2-CN-P-1A 2-CN-P-1C	No Load Shedding	1-FW-P-1B 1-FW-P-1C 1-CN-P-1B 1-CN-P-1C 1-FW-P-1A 2-FW-P-1B 2-CN-P-1A 2-CN-P-1B

The load shedding scheme may be administratively defeated by means of a two position load shedding defeat switch. The two positions are "Normal" and "Defeat". Both this switch and the unit startup switch are key operated switches which are located on the Unit 1 A Station Service Transformer panel (1-EP-CB-03A) which is located in the Unit 1 emergency switchgear room.

If the load shedding switch is in "Defeat" or there is a loss of control power to the load shedding control circuit, these situations will be annunciated in the control room (windows 1J-32 and 2H-61).

When both units are operating at power, the load shed switch can be in either "Normal" or "Defeat." Following a two unit trip with the switch in "Normal", all feedwater and condensate pumps for both units would be tripped. The auxiliary feedwater pumps would start to provide a safe shutdown of both units. If a two unit trip occurred with the switch in "Defeat", both the emergency and normal buses would experience reduced voltage. Subsequently the emergency buses would be taken off the RSS system and energized by the emergency diesels as a result of the degraded voltage protection scheme. Voltage should then return to a nominal level on the normal station service buses.

In situations where one unit is at power and the other unit is shutdown for a prolonged period (refueling or maintenance outage) and the shutdown unit's load on the RSS is small, the load shed switch should be in "Defeat". Should the operating unit trip under these conditions, the voltage on the RSS system should be acceptable.

SUMMARY OF SAFETY ANALYSIS: The RSS power system load shedding does not constitute an "unreviewed safety question" as defined in 10CFR 50.59. The load shedding removes non-safety related, secondary plant, electrical equipment loads from the RSS system under potentially high RSS transformer load conditions. The result of the load shedding is to improve the voltage profile on the RSS system. The load shedding will not trip or prevent automatic starting of any equipment important to safety, nor does it revise protection or logic schemes of any equipment important to safety previously evaluated in the safety analysis report.

The RSS load shedding modification could, under certain conditions, cause the normal feedwater pumps to be tripped and prevented from automatic starting. However this type of accident has been previously evaluated in the FSAR, Section 15.2.8 "Loss of Normal Feedwater", and the conclusions remain unchanged. All new components and materials used for modification of nonsafety related equipment are totally compatible (relative to design, quality, and functional requirements) with original components and materials. Therefore, the possibility for a malfunction of a different type than previously evaluated does not exist.

The modification does not reduce the capacity, method of operation, or design basis for any safety related systems or component for any of the postulated accidents. Therefore, this modification does not reduce the margin of safety as defined in any technical specification.

35. DC 79-S48 Reducing Nozzle Loads on Reactor Coolant Pump Motor Coolers

Description: The component cooling water piping and associated supports to the reactor coolant pumps (RCP) were reanalyzed for the effect of increased service water and component cooling water temperatures. It was determined that the loads placed on the nozzles of the RCP motor stator and lube oil coolers exceeded Westinghouse limits.

The resolution was to install flanged sections of flexible stainless steel hoses in the CC lines where they connect to the stator and lube oil coolers for each of the reactor coolant pumps.

SUMMARY OF SAFETY ANALYSIS: Reducing RCP nozzle loads by adding Flexible Metal Pipe does not create an "unreviewed safety question", as defined in 10 CFR 50.59.

The original design basis of the component cooling water system has not been changed. The level of integrity of the component cooling water system is not changed.

36. DC 79-S50 Replacement of Containment Isolation Solenoid Valves

Description: The solenoid valves installed to control six air operated valves (TV-SS100A, TV-SS101A, TV-SS102A, TV-SS104A, TV-SS106A, and TV-SS112A) are located inside the containment and are part of the containment isolation system. During a review of solenoid operated valves for compliance to IE Bulletins 79-01 and 79-01A, it was determined that the solenoid operated valves (SOV's) on these six valves had no qualification data for a post-accident environment. As a result, the SOV's were replaced with qualified models.

SUMMARY OF SAFETY ANALYSIS: Replacement of the solenoid valves does not create an "unreviewed safety question" as defined in 10 CFR 50.59. The replacement solenoid valves have been qualified to the North Anna Unit 1 post-accident environment. Therefore, this work does not change the plant design.

37. DC 79-S51 Fuel Pool Cooler Support Modification

Description: The service water system was originally designed for a maximum service water reservoir temperature of 95° F. Recent analysis of multi-unit operation during accident conditions indicates that the service water reservoir temperature may exceed 95° F. Therefore, the structural design of the support frame for the fuel pool coolers was reevaluated.

Stress anlaysis using a service water reservoir temperature range of 30° F to 115° F was completed to determine what changes were to be made to the supports. As a result, steel reinforcing plates were added to the embedments in the pool cooler support structures in order to provide additional shear capability.

SUMMARY OF SAFETY ANALYSIS: The Stress Analysis on service water piping and associated fuel pool coolers supports does not create an "unreviewed safety question" as defined in 10 CFR 50.59. The original design basis of the Service Water System have not been changed. The level of integrity of the Service Water System is not changed.

38. DC 79-S53 Nitrogen Supply to Power Operated Relief Valves

Description: The motive power for the pressurizer power operated relief valves PCV-1455C and PCV-1456 was supplied by the non-safety grade containment instrument air system. In order to comply with NUREG-0578, the addition of a separate safety grade system became necessary to provide a redundant source of motive power to the PORV's during normal plant operation. A nitrogen supply system (see DC-78-44) was previously provided to supply motive power to the PORV's during solid water operation. This system was modified to satisfy the concerns of the NRC.

A new 3/4" stainless steel supply line was tied into the nitrogen supply line downstream of the pressure control valves and terminated in an existing 3/4" containment instrument air line upstream of SOV-1455C-1. Check valves were added in the air and nitrogen lines to prevent the supply piping from being depressurized in the event of a loss of containment air or a loss of the nitrogen supply system.

SUMMARY OF SAFETY ANALYSIS: The nitrogen supply to the PORV's modification does not create an "unreviewed safety question" as defined by 10 CFR 50.59. This modification will help insure automatic operation of the PORV's by providing a safety grade motive power should the containment instrument air supply be lost.

The modified containment air system tubing exceeds original design requirements and does not alter the normal operation of the PORV.

The modified portions of the system exceed original design requirements for containment instrument air. The modification should result in increased reliability of the pressurizer PORV's and does not alter system characteristics of pressure relief system or any other safety related system.

The margin of safety as defined in the basis for any Technical Specification is not reduced as this modification has no affect on the Technical Specifications.

39. DC 79-S56 Auxiliary Feedwater System - Valve Position and Flow Indication

Description: During normal operation, the discharge of the auxiliary feedwater pumps is lined up to supply water to the steam generators as soon as the pumps start and build up sufficient discharge pressure. MOV-FW100B and D and HCV-FW100C are left open while MOV-FW100A & C and HCV-FW100A & B are closed to assure adequate feedwater flow to each of the steam generators. PCV-FW159A and B which are designed to maintain adequate discharge header pressure in order to prevent running out the motor driven auxiliary feed pumps, are normally closed and automatically regulate open upon starting of the pumps. All of the above mentioned valves do not receive automatic signals to go to their required positions upon initiation of the auxiliary feedwater system, and the control room operator does not have any annunciation to alert him if any valves are out of their required position.

Another concern as delineated in NUREG-0578 concerns the reliability of auxiliary feedwater flow instrumentation. The design for this instrumentation must be able to satisfy the single failure criterion. The "fall-back" position has been adopted in this case for satisfying this criterion in that flow indication will be backed up with steam generator wide range level indication. This was accomplished by changing the vital bus power feeds to the flow instrumentation such that the associated flow and level instruments for each steam generator are fed from diverse vital power supplies. The power supplies for the auxiliary feed pump suction pressure instruments (PT-FW-103A, B, C) were exchanged with the power supplies to the flow instruments (FT-FW100A, B, C).

In order to provide assurance that the auxiliary feedwater system is properly aligned, several annunciators were added in the control room to alert the operator when a valve was not properly positioned. Two annunciators (windows 1A-6 and 1A-14) monitor the status of the HCV's. To make sure that PCV-FW-159A, B open properly after the auxiliary feedwater pumps start, an annunciator (window 1A-19) was added which alerts the control room operator if the PCV's are not open 20 seconds after the pumps start. The annunciation was delayed 20 seconds after pump starting to avoid having the annunciator locked in while the pumps were not operating and to avoid spurious alarms while the pumps were coming up to speed.

SUMMARY OF SAFETY ANALYSIS: Installation of AFWS discharge valves position annunciation and upgrading the AFWS flow indication does not create an "unreviewed safety question," as defined in 10 CFR 50.59. This modification increases the system reliability.

The modification will not affect automatic or manual starting of the AFWS, valve positions, nor does it revise protection or logic schemes of any equipment important to safety previously evaluated in the safety analysis report.

The modification does not affect the functioning nor position of any valves and it provides a more reliable power source for flow indication to the steam generators from the AFWS. Therefore, the possibility for an accident of a different type than evaluated in the FSAR is not created.

The modification does not affect system capacity, method of operation, or design basis for any safety related components or systems for any postulated accidents. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

40. DC 79-S57 Main Steam Valve House Pipe Support Modifications

Description: The main steam lines were supported by gapped restraint/stops next to the main steam containment penetration exterior to the containment. The purpose of the gapped stop was to allow thermal growth but prevent excessive downward movement of the main steam piping while the main steam safety valves lift. Excess movement downward because of too large a gap could have caused overstressing of the containment penetration. Contact of the main steam piping with the restraint during periods of normal heatup because of too small a clearance could have resulted in the buildup of stresses in excess of existing structural analysis. It was difficult to assure the required gap under all normal operating conditions.

The resolution was to substitute a non-gapped restraint for the existing gapped restraint/stop. Based on the resulting revised stress analyses, several modifications to hangers, restraints, and suppressors were required in order to redistribute loads in the piping system to keep stresses within allowable ranges based on original design criteria.

SUMMARY OF SAFETY ANALYSIS: The Main Steam Valve House Pipe Support Modification does not create an "unreviewed safety question" as defined in 10 CFR 50.59.

This modification does not change the level of integrity of the main steam piping for the design basis for the pipe supports. The original design basis of the system and its supports is not changed.

41. DC 79-S58 Steam Generator Water Level Reference Leg Insulation

Description: High energy line breaks inside the containment could result in the heatup of the steam generator level instrument reference leg. Increased reference leg water temperature would result in a decrease of the water column density with a consequent apparent increase in the indicated S/G narrow range level above actual level. This potential level bias could result in delayed generation of protection signals for reactor trip and auxiliary feedwater initiation which are initiated by low-low steam generator water level.

The resolution was to insulate the upper tap reference leg and lower tap instrument tubing for all three narrow range level transmitters on each steam generator. In addition, the steam generator low-low level trip setpoint was increased form 5% to 15% (narrow range). Further evaluations as the the effectiveness of these changes may result in future changes to the low-low level setpoint.

SUMMARY OF SAFETY ANALYSIS: UNREVIEWED SAFETY QUESTION: The addition of insulation to the steam generator narrow range water level reference legs does not create an "unreviewed safety question" as defined in 10 CFR 50.59. The intent of this change is to assure that protection signals (reactor trip and auxiliary feedwater initiation), which are based on low-low steam generator water levels, are not delayed. Therefore, this change decreases the probability of occurrence of the consequences of an accident or malfunction of equipment important to safety, previously evaluated in the safety analysis report.

The change decreases the possibility of delayed protection signals due to an adverse containment environment.

The margin of safety as defined for any technical specification is not reduced as it has no effect on the technical specifications.

42. DC 79-S62 Diesel Generator Cooling System Modification

Description: The emergency diesel is cooled by circulating coolant through radiators that are cooled by a cooling fan which circulates air through the radiators. The pitch of the fan blades is adjustable. The cooling fan is driven from the power takeoff shaft which is coupled to the diesel's lower crankshaft. Seasonal fan blade pitch settings had been required because of variations in ambient air temperature which change the torsional load on the fan drive shaft. The limiting component was the fan drive shaft coupling with a rating of 180 H.P. Experience had shown that the seasonal changes to the fan blade pitch were impractical because of the difficulty in adjusting the blade pitch.

The resolution involved a modification to the drive shaft coupling by changing the four bolt configuration of the crankshaft and power takeoff shaft flanges to a six bolt configuraiton. This modification increased the rating of the coupling from 180 H.P. to 230 H.P. which is sufficient to eliminate the need for seasonal changes to the fan blade pitch.

The need for seasonal changeout of the coolant still exists. A corrosion inhibitor is added to the coolant the year round, and ethyl glycol (used for freeze protection) is added during the winter months. Plans are to drain the ethyl glycol coolant mixture to the cooling system sump during the summer months and fill the coolant system with water and corrosion inhibitor. Then on the onset of colder weather, the water-glycol-inhibitor solution will be pumped from the sump into the coolant system after the water with corrosion inhibitor is dumped to the drain system. This will greatly reduce the expense of the seasonal changeout of coolant.

SUMMARY OF SAFETY ANALYSIS: The diesel generator fan drive shaft modification does not create an "unreviewed safety question", as defined by 10 CFR 50.59. This modification should insure more reliable operation of the emergency diesel generator through a variety of operating conditions by eliminating the need for periodic adjustment of the cooling fan blade pitch settings.

The modified emergency diesel generator cooling system exceeds original design requirements and does not change intended function reliability or operation of the emergency diesel generator.

The modification should result in increased reliability of the emergency diesel generator by reducing maintenance on the diesel cooling system and does not alter the intended purpose of the emergency diesel generator or any other safety-related system or component.

The margin of safety as defined in the basis for any technical specification is not reduced as this modification has no effect on the technical specifications.