1995 ANNUAL REPORT

MOD 85-3-066 CVCS, REV. 1

CVCS HOLDUP TANKS VACUUM PROTECTION

Descripion and Purpose

The purpose of this Nuclear Safety Evaluation (NSE) was to evaluate the installation of pressure sensing devices and actuation circuitry for the Chemical Volume and Control System (CVCS) Holdup Tanks (HUTs) to protect them against implosion. On a decreasing HUT pressure condition approaching atmospheric, the three pressure switches are designed to trip any of the three associated HUT pumps.

Summary of Safety Evaluation

The above mentioned instruments, actuation circuitry, and associated pumps do not provide any safety-related functions. This modification reduces the probability of a radioactive release from the HUTs. The changes do not create the possibility of an accident or malfunction of a different type than any previously evaluated in the FSAR.

1995 ANNUAL REPORT

NSE-86-03-150 SIS, REV.0

ELIMINATION OF BORON INJECTION TANK, PHASE I

Description and Purpose

The purpose of this evaluation was to analyze the first of a two phase elimination of the Boron Injection Tank (BIT). Phase I eliminated the high concentration of boric acid from the BIT and the retiring in place of the BIT heaters and heat tracing on the interfacing piping with the BIT. A Technical Specification Amendment associated with this safety evaluation has been submitted.

Summary of Safety Evaluation

The modification does not increase the probability or possibility of occurrence of an accident or malfunction of safety related structures, systems or components previously evaluated in the FSAR. The changes involved existing systems and do not have un-analyzed effects on the ability to mitigate the consequences of postulated accidents. The functional elimination of the BIT does not involve a significant reduction in the margin of safety involving the Containment Integrity and Core Response Analyses.

1995 ANNUAL REPORT

NSE 87-3-066 HV, REV. 0

REPLACEMENT OF UPPER AND LOWER BYPASS DAMPERS FOR FUEL STORAGE BUILDING EMERGENCY EXHAUST SYSTEM

Description and Purpose

This Nuclear Safety Evaluation (NSE) addressed the replacement of the upper and lower bypass dampers for the Fuel Storage Building (FSB) Emergency Exhaust System with plates, to ensure that the leakage requirements of the Fuel Storage Building ventilation system dampers are met in the unlikely event of a radioactive release.

Summary of Safety Evaluation

The replacement of the upper and lower bypass dampers of the FSB emergency exhaust system with the specially fabricated hinged plates virtually eliminates any leakage through the damper assemblies thereby ensuring compliance with Section 3.8 of the IP3 Technical Specifications. The plates will be closed only during refueling operations and at any other time that the FSB ventilation system is required to be operable in accordance with Section 3.8 of the Technical Specifications. The installation of a "manual" damper assembly ensures that the bypass dampers are sealed in the incident mode.

1995 ANNUAL REPORT

NSE 88-3-112 COND, REV. 3

REDUCTION OF DISSOLVED OXYGEN IN THE CONDENSATE STORAGE TANK

Description and Purpose

This Nuclear Safety Evaluation (NSE) addressed the use of a nitrogen blanketing system to improve the ability of the Condensate Storage Tank (CST) to maintain dissolved oxygen levels in the water within the EPRI recommended concentration range of 5 to 100 parts per billion (ppb) of water. This safety evaluation has resulted in a change to the basis of the Technical Specifications.

Summary of Safety Evaluation

Installation of this modification increased the ability to maintain low oxygen levels in the stored water in the CST. The low oxygen content resulted in improved chemistry control of feedwater to the steam generators. Failure of this system will not adversely affect the ability of the CST to perform its safety related functions.

1995 ANNUAL REPORT

MMP 89-3-238 WDS, REV. 1

REACTOR COOLANT DRAIN TANK PUMP REPLACEMENT

Description and Purpose

This minor modification (MMP) replaced the Reactor Coolant Drain Tank Pumps (RCDT pumps) with a more reliable one (RCDT pump replacement was reported in the 1994 Annual Report) and also removed the low discharge pressure trip signal and the associated time delay relays. This trip function protected the pumps against cavitation due to clogging of the strainers located in the suction line of each pump. This trip function is no longer required since the strainers have been removed and the new pumps are cooled by air rather than water.

Summary of Safety Evaluation

The replacement of the Reactor Coolant Drain Tank Pumps will allow the liquid waste disposal system to operate as originally designed. The new RCDT pumps are designed to accommodate the process fluid and will result in improved waste transfer and minimize waste flow into the Containment Building sump.

1995 ANNUAL REPORT

NSE 90-3-036 RMS, REV. 0

IP-3 RADIATION MONITORING SYSTEM UPGRADE - PHASE II
CONDENSATE POLISHER EFFLUENT RADIATION MONITOR - R61 UPGRADE

Description and Purpose

This modification relocated the sampling point of radiation monitor R61 to the discharge of the Total Disolved Solids (TDS) Waste Collection Pumps. Sample valves have been added to monitor radioactivity in the liquid released to the discharge canal from the High and Low (TDS) tanks. A high radiation alarm, recorder and radiation monitor controller with indicator has also been added to the Central Control Room to allow for continuous recording functions.

Summary of Safety Evaluation

This modification does not increase the probability of an accident or malfuntion of equipment important to safety becauses no changes were made to the detection and monitoring requirements of R61 and the modification is an improvement over the previous design because it allows for selective sampling of either the HTDS or the LTDS tanks during recirculation and discharge.

1995 ANNUAL REPORT

MOD 90-3-124 MBFP, REV. 1

MAIN BOILER FEED PUMP OIL ACCUMULATORS

Description and Purpose

The purpose of this modification is to reduce the number of inadvertent plant trips at IP-3 caused by the Main Boiler Feed Pump (MBFP) Turbine Control/Lube Oil System by installing two 80 gallon oil accumulators in the high pressure control oil system. The addition of these accumulators stabilizes system pressure to preclude sudden transients from tripping the MBFP Turbine, before the back-up main oil pump has had an opportunity to start. Additionally, automatic fire suppression was added for the areas containing the new oil accumulators and piping.

Summary of Safety Evaluation

The changes and/or additions made under this modification have no impact on, nor are they located in the proximity of any nuclear safety-related equipment, therefore they will have no effect on the FSAR. The addition of the MBFP accumulators does not have any safety function. However, by helping to eliminate potential unit trips, the frequency of challenges to the Reactor Protection System is reduced.

1995 ANNUAL REPORT

MMP 90-3-220 SI, REV. 0

REMOVAL OF GEMS LEVEL COLUMNS LT-938, 939, 940 AND 941

Description and Purpose

The purpose of this modification was to remove Gems level columns LT-938, LT-939, LT-940 and LT-941 and temporary retirement of their Control Room indicators LI-938, LI-939, LI-940 and LI-941. They were not accurate, and their task is performed by analog transmitters LT-1251, 1252, 1253, 1254, 1255, and 1256 previously installed through Modification 80-3-052 ESS.

Summary of Safety Evaluation

The removal of the GEMS level transmitters (LT-938, 939, 940, and 941) was acceptable since the function of monitoring Containment Building water level is still intact through other EQ transmitters previously installed. The NRC Regulatory Guide 1.97 requirements are still maintained, and the margin of plant safety was not reduced. This modification will not increase the probability of an accident.

1995 ANNUAL REPORT

NSE 91-3-027 SP, REV. 1

POST ACCIDENT H2 MONITORS REPLACEMENT

Description and Purpose

This modification replaced two redundant hydrogen monitors (Train A and B) with two improved monitors that are capable of performing their design function with the high moisture that is carried through the sample lines.

Summary of Safety Evaluation

The original monitoring system was subject to mis-operation due to a high amount of condensate in the sample lines. The new monitors can operate with condensate in their detector units and provide more reliable information.

The modification will improve the conatinment hydrogen monitoring reliability since any moisture carry-over will not adversely affect the analyzer performance. This modification will simplify the hydrogen monitoring system by eliminating the need for a reagent gas.

1995 ANNUAL REPORT

NSE 91-3-046 CVCS, REV. 0

CHARGING PUMPS PIPING VIBRATION

Description and Purpose

This modification installed tie-back supports between the vent/drain piping and the main headers of charging pumps 31, 32 and 33 to reduce stresses caused by vibration.

Summary of Safety Evaluation

The new supports are an improvement which will increase the fatigue life of the system. The siesmic and pressure integrity, along with the operation of the charging pumps, is not compromised by the upgrading of the pipe supports.

1995 ANNUAL REPORT

MMP 91-03-118-ED, REV. 0

AUXILIARY BOILER FEED PUMP ROOM CABLE TRAY FIRE BARRIER INSTALLATION

Description and Purpose

The purpose of this modification was to enclose the 12 inch horizontal tray at elevation 18'6" in the Auxiliary Feed Pump Building to meet the existing cable tray separation criteria.

Summary of Safety Evaluation

The original design criteria was met with the design requirements of this modification. The installed tray cover performs the same function as a required vertical barrier. The Fire Protection System was not degraded since the covered tray is equivalent to the original vertical barrier and in compliance with the original plant design as stated in the plant Electrical Separation Criteria document.

1995 ANNUAL REPORT

MMP 91-3-199 H_2, REV. 0

REPLACEMENT OF THE GENERATOR HYDROGEN PURITY/PRESSURE MONITOR

Description and Purpose

This minor modification (MMP) replaced the Purity/Pressure Monitor, in the Generator Hydrogen Control Panel located on the 15 foot elevation of the Turbine Building, with an upgraded model recommended by Westinghouse. The previous monitor and its components are no longer manufactured by Westinghosue.

Summary of Safety Evaluation

The upgraded system replaced the previous mechanical-pneumatic operated parts with electronic components which are easier to calibreate, more accurate and has fewer mechanical parts to maintain. The replacement monitor meets the criteria for monitoring the generator's hydrogen gas and is not safety significant.

1995 ANNUAL REPORT

MMP 92-3-040 LWD, REV. 0

REPLACEMENT OF FC-1002 (OLD)

Description and Purpose

This minor modification (MMP) replaced the existing Controlotron 241R flow computer (FC-1002) with a new Controlotron Series 990 System (FI-1002, FE-1002). The flow computer monitors instantaneous and total flow of water from the Vapor Containment (VC) sump pump discharge line to the waste holdup tank.

Summary of Safety Evaluation

This modification installed a more accurate system to measure and record the total amount of water sent to the waste holdup tank from the containment sump through line 338. The system provides operators with the instantaneous flow as shown on flow recorder FR-1024 on flight panel FD, and locally at the waste disposal panel the computer provides total flow and instantaneous flow.

1995 ANNUAL REPORT

MMP 92-3-164 MULT, REV. 0

PLANT ELECTRICAL SYSTEM FUSE UPGRADE

Description and Purpose

This modification corrected discrepancies and inconsistencies with fuse sizes in MCC's No. 32, 33, 34, 36A, 36B and MCC-37 in the 480V system.

Summary of Safety Evaluation

The replacment fuses satisfy the requirements of sizing to allow the load to perform its function, protection of the conductor serving the load, and coordinate with other protective devices of the circuit.

1995 ANNUAL REPORT

MOD 92-3-251 IA, REV. 0

INSTRUMENT AIR DRYER AND PIPING REPLACEMENT

Description and Purpose

This modification replaced the existing desiccant and refrigerant dryers and after-filters with one new heatless desiccant dryer which included a pre-filter and after-filter set, and upgraded and relocated the existing temporary heatless dryer to a permanent position. In addition, dew point monitors, flow measurement equipment and a low pressure alarm in the Turbine Building were installed.

Summary of Safety Evaluation

The new dryers improved the performance characteristics of the Instrument Air System. The service life of the Instrument Air System is extended by this modification.

1995 ANNUAL REPORT

MOD 92-3-270 DC PWR, REV. 1

ADDITIONAL 125VDC STATION BATTERY

Description and Purpose

This modification was implemented to alleviate the load management of the 480V Motor Control Centers (MCC's) during peak accident loading scenarios, when 480 buses are from either Offsite Power or Emergency Diesel Generator or Appendix R Diesel Generator. To accomplish this, a new 125VDC non-safety Station Battery 36 together with all the necessary auxiliaries and support systems was installed.

Summary of Safety Evaluation

The installation of Battery 36, Battery Charger 36, DC Power Panel 36, Heating and Ventilation Systems, accessories does not increase the probability of occurrence or consequences of an accident or malfunction of structures, systems or components important to safety previously evaluated in the FSAR since the modification enhances 480V AC Load Management by providing additional power supply to alleviate the overload concerns. FSAR Table 8.2.2, FSAR Figures 8.2-6 and 10.2-46 have been revised to reflect the changes covered by this MOD.

1995 ANNUAL REPORT

NSE 93-3-142 SWS, REV. 2

EVALUATION OF TYING OR CROSS CONNECTING THE SERVICE WATER SYSTEM HEADERS AT COLD SHUTDOWN

Description and Purpose

This Nuclear Safety Evaluation (NSE) evaluates the safety significance of allowing the essential and the non-essential Service Water System (SWS) Headers to be tied together when in cold shutdown. This revision restores the throttling requirements that had been specified in Rev.0 of this NSE, but which had been removed by Rev.1.

Summary of Safety Evaluation

The procedure generated by this NSE affects SWS alignment at CSD only, and neither challenges the design limits of any Service Water System component, nor impacts any system subject to operational transients or design accidents. It was concluded that tying of the service water system headers at Cold Shutdown does not present an unreviewed safety question.

1995 ANNUAL REPORT

MMP 93-3-159 EX, REV. 0

REPLACEMENT OF 5TH EXTRACTION STEAM BRANCH CONNECTIONS, FITTINGS, AND ASSOCIATED PIPING

Description and Purpose

This modification replaced four branch connections in the 5th Extraction Steam system, which were not designed in accordance with applicable sections of the "Power Piping" code (B31.1). In addition, because of pipe wall thinning, dimensional interferences, and other problems, several fittings and sections of pipe were replaced. Because of concerns about erosion/corrosion in this system, the piping and fittings used in the replacement were fabricated from chrome-moly steel, which is a vast improvement over the carbon steel components in terms of erosion/corrosion resistance.

Summary of Safety Evaluation

The replacement of 5th Extraction Steam piping and fittings was a Non-Category I undertaking, and did not impact the functioning of any safety-related systems. The new material used greatly improved piping resistance to erosion/corrosion as well as the possibility of a high energy steam line rupture, and did not in any way degrade the functioning of the Extraction Steam system. No unreviewed safety questions are presented by the performance of this modification.

1995 ANNUAL REPORT

MMP 93-3-165 COND, REV. 0

REMOVAL OF FCV-1150 OPENING FEATURE

Description and Purpose

The purpose of this modification was to permanently disconnect the control circuit for the condensate bypass valve FCV-1150. The automatic opening function of FCV-1150 was disconnected by Temporary Modification TM-1357 and the valve was mechanically locked closed. The permanent closure of FCV-1150 had previously been evaluated by NSE 82-3-080 COND.

Summary of Safety Evaluation

The automatic bypass function provided by valve FCV-1150 did not perform satisfactorily in helping to stabilize plant operation during a 50% load rejection transient. In addition, the design of the control circuitry for FCV-1150 could spuriously actuate, thus creating a transient load. NSE 82-3-080 COND was prepared and approved to remove FCV-1150 from plant operation. It was concluded that this modification would not involve an unreviewed safety question, does not require a change in the Technical Specifications. However, FSAR Figures 10.2-3A, 8.2-6, and 8.2-9 were revised to show the removal of the control circuitry of valve FCV-1150.

1995 ANNUAL REPORT

MMP 93-3-222 EX, REV. 0

REPLACEMENT OF 6TH EXTRACTION STEAM BRANCH CONNECTION AND PIPING; AND ADDITION OF PADS ON 5TH AND 6TH EXTRACTION STEAM BRANCH CONNECTIONS

Description and Purpose

This modification replaced one lateral branch connection and added reinforcement pads on six tee connections in the 5th and 6th Extraction Steam systems. These connections were not designed in accordance with applicable sections of the ANSI B31.1. (1967) "Power Piping" code. In addition, because of pipe wall thinning, a section of pipe adjacent to the lateral connection was also replaced. Because of availability, the fitting used in replacement was fabricated from chrome-moly steel, which was an improvement over the existing carbon steel component in terms of erosion resistance. The section of replacement piping was carbon steel, which matched the existing extraction steam piping.

Summary of Safety Evaluation

The replacement of a 6th Extraction Steam piping fitting and addition of pads on the branch connections in the 5th and 6th Extraction Steam Piping is a Non-Category I undertaking and did not impact the functioning of any safety-related systems. The new lateral and added pads improved piping resistance to erosion as well as to the possibility of a high energy steam line rupture, and did not degrade the functioning of the Extraction Steam system. No unreviewed safety questions were presented by this modification.

1995 ANNUAL REPORT

MMP 93-3-251 SWS, REV. 0

TEMPERATURE CONTROL VALVES (TCV'S) INSTALLATION

Description and Purpose

The purpose of this modification was to install temperature control water flow regulating valves on the Service Water System outlets of the 31 and 32 Central Control Room Air Conditioner (CCRAC) unit condensers. The installation of these valves will prevent evaporator icing and allow the CCRACs to operate more efficiently.

Summary of Safety Evaluation

The installation of the four temperature control water flow regulating valves will enhance the overall operability of the CCRAC Units 31 and 32 and will not in any way negatively impact any safety related system. The failure of one valve in the closed position will not stop the flow of cooling water through all the condensers. Therefore, no unreviewed safety question are presented by the performance of this modification. This modification did require a change to a drawing described in the FSAR.

1995 ANNUAL REPORT

MOD 93-3-256-480V, REV. 2

ALTERNATE POWER FEED TO PAB EXHAUST FAN 32 FROM 480V MCC-312A

Description and Purpose

This modification was to help plant operators alleviate the load management of 480V Bus 6A during emergency condition by reducing load on the bus. The transfer of power feed for PAB Exhaust Fan 32 from normal source 480V Bus 6A to alternate source MCC-312A was carried out by operating a Non-Automatic Transfer Switch whenever required. A Kirk Key Interlock was provided to allow the transfer only when both power supplies (normal or alternate) to the fan are de-energized. The fan will be manually controlled from Fan Room Control Cabinet located in the PB Fan Room at Elevation 80'-0".

Summary of Safety Evaluation

This safety evaluation reviewed the potential safety impact of providing an alternate power feed for the Primary Auxiliary Building Exhaust Fan 32 from 480V MCC-312A.

1995 ANNUAL REPORT

MMP 93-3-261 CWM, REV. 0

PERMANENT CITY WATER TO CLOSED COOLING WATER HEAT EXCHANGERS

Description and Purpose

This minor modification (MMP) permanently piped city water to the Turbine Hall Closed Cooling Water (THCCW) Heat Exchanger. City water is used as an alternate cooling when service water is not available. This occurs during "loss of offsite power," safeguard initiation signals, and other site emergencies where condensate pump motor gearing cooling is required. The scope included piping from the city water valve to a backflow preventer valve to the two heat exchangers along with the required supports.

Summary of Safety Evaluation

The installation of a permanent city water line to the THCCW Heat Exchangers was a QA Non-Category I undertaking which did not affect any nuclear safety-related systems. No unreviewed safety questions were presented by performing this modification. Detailed design bases for this conclusion are found in Section 5.0 ("Technical Evaluation") of the MMP.

1995 ANNUAL REPORT

MMP 93-3-270 ARM, REV. 1

MODIFICATION OF AREA RADIATION MONITOR R-10

Description and Purpose

The purpose of this modification was to allow plant personnel to achieve a more accurate low end calibration and read out of radiation levels. To accomplish this a new pump circuit was installed on the log pump board and new radiation level indicators were installed in the Central Control Room and at the R-10 Radiation Monitor, (ABFP building area monitor).

Summary of Safety Evaluation

This modification did not introduce any new potential failure mechanisms to the radiation monitoring system or to any safety related system in the plant. It did not change the technical specifications. This modification improved the detector by increasing the sensitivity of the scale therefore providing more accurate readings.

1995 ANNUAL REPORT

MMP 93-3-333 SWS, REV. 0

INSTALLATION OF SERVICE WATER HEADER LOW POINT DRAIN FOR INTAKE STRUCTURE

Description and Purpose

The purpose of this minor modification (MMP) was to provide a method of freeze protection for the 8" Service Water Header supplying the circulating water pump seals/bearings, through the design and installation of a low point drain.

Summary of Safety Evaluation

This modification affects the Non-Category I, Seismic Class III portion of the Service Water System only. The safety related portion is not affected physically or functionally. This minor modification will upgrade and increase the Service Water System's reliability during winter outages by providing a method of draining the piping and thus affording the line a form of freeze protection. This modification did not affect this system's, nor any other safety related plant system's ability to function during normal or accident conditions. The compatibility of materials and pressure Class utilized for the drain components with that of the SW System, ensures that there are no unreviewed safety question with this installation.

1995 ANNUAL REPORT

MMP 93-3-363 COND, REV. 2

CST LEVEL SWITCH REPLACEMENT

Description and Purpose

This modification revised the level control scheme for the Condensate Storage Tank (CST). The original level switches did not have sufficient accuracy to maintain the minimum level in the CST of 360,000 gallons as required by Technical Specifications without infringing on tank level during normal plant operation. This was due to a potential +/- 10% setpoint shift during and following a seismic event which must be accounted in the setpoint, placing the setpoint alarm and tank isolation into normal operating region so that nuisance alarms and/or inadvertent action of the CST isolation valves will be avoided.

Summary of Safety Evaluation

This modification installed a more accurate system to monitor minimum water level in the Condensate Storage Tank required to maintain the plant in hot shutdown condition for 24 hours following a reactor trip from full power. The new design did not change the operation logic of isolation valves and setpoints for valve closure and alarm actuation. This modification enhances the function of the level control logic by improving the level control accuracy. It is therefore concluded that this modification did not involve an unreviewed safety question.

1995 ANNUAL REPORT

MOD 93-3-419 CRHV, REV. 1

ADDITIONAL CCR HVAC COOLING

Description and Purpose

This modification increased the normal cooling capacity of the Central Control Room (CCR) HVAC system. This modification implemented ability to maintain 75°F in the CCR during normal operation with the outside air temperature at 93°F and the cooling water inlet temperature at 95°F.

Summary of Safety Evaluation

This mondification installs indepentent, non-safety-related, A/C equipment to supplement cooling capacity for the CCR during normal operation. Supplemental cooling provided by the new A/C units in the CCR ensures that the environmental conditions for supporting normal operation are maintained. The operation of the supplemental cooling equipment will not affect any other safety-related SSC in such a way that this modification could be considered an accident initiator. This activity ensures that existing boundaries are maintained with respect to existing fire, radiation dose, and security requirements, thereby precluding the possibility of a new accident.

1995 ANNUAL REPORT

MMP 93-3-429 480V, REV. 1

RELOCATION OF POWER FEED TO SWITCHGEAR ROOM EXHAUST FAN 33

Description and Purpose

This modification addressed the single failure concern associated with Switchgear Room Exhaust Fans 33 and 34 by relocating the power feed from 480V MCC-39 to 480V MCC-36A and MCC-36C respectively, so that the fans continue to operate, without any manual load management, following a design basis accident and/or loss of offsite power. Previously, MCC-39 supplied power to both Switchgear Room Exhaust Fans 33 and 34, and MCC-39 required manual load management of the selected non-safeguard loads (including Switchgear Room Exhaust Fans 33 and 34) during peak accident loading scenarios. Modification 93-3-257 480V relocated the power for Fan 34 from MCC-39 to 480V MCC-36C.

Summary of Safety Evaluation

This safety evaluation reviewed the potential safety impact of relocating the 480V power feed and the 120V control power feed for the Switchgear Room Exhaust Fan 33. Based on the review and analysis above, it was concluded that enhanced reliability of the safeguards electrical equipment resulted from this modification.

1995 ANNUAL REPORT

MMP 94-3-003 WCCPPS, REV. 0

AIR RECEIVER PRESSURE SWITCH REPLACEMENT

Description and Purpose

This minor modification (MMP) replaced pressure switches (PC-1303S, PC-1304S, PC-1305S, and PC-1306S) on air receiver 31, 32, 33, and 34. This MMP also raised the minimum air receiver pressure from 80 psig to a new analytical limit of 90 psig, sets the switches low pressure setpoints to 93.5 psig.

Summary of Safety Evaluation

The pressure switches are not described in the FSAR. However, the setpoint is described in the FSAR, Figure 6.6-1. Although the Weld Channel Containment Penetration Pressurization System (WCCPPS) is an engineered safety feature, no credit is taken for its operation in calculating the amount of radioactivity released for offsite dose evaluations. Changing the switch hardware and the minimum air receiver pressure setpoint has been determined not to create an unreviewed safety question.

1995 ANNUAL REPORT

MMP 94-3-009 CRHV, REV. O

CCR HVAC BACKUP NITROGEN SUPPLY

Description and Purpose

This minor modification (MMP) installed a backup gas supply to the pneumatic actuators which position dampers in the Central Control Room (CCR) HVAC System. Licensee Event Report (LER) #93-045-00 identified a design deficiency associated with a loss of function of the Instrument Air System. Under such a scenario, some of the HVAC dampers would fail in a position that would result in a loss of HVAC system functionality. This, in turn, could adversely affect CCR habitability and equipment functionality.

The backup gas supply will function automatically and for a prolonged period of time should the Instrument Air System become inoperable for any reason. This will permit the pneumatic actuators to properly position the dampers for any mode of system operation.

Summary of Safety Evaluation

The damper pneumatic actuators are required to be functional during the three possible modes of CCR HVAC System operation, which are; normal, 10% incident and 100% recirculation. By providing a passive automatic backup gas supply, continued functioning of the damper actuators after a loss of Instrument Air supply pressure is assured. Sizing of the backup gas supply was based on having available, at all times, a minimum of 24 hours of pressurized gas to cycle the actuators and to compensate for conservatively determined component leakage. After installation, testing will be conducted using approved procedures to ensure that the installed equipment performs as designed and that interfacing equipment is not adversely affected.

1995 ANNUAL REPORT

MMP 94-3-042 IA, REV. 0

MS-PCV-1139 SOV UPGRADE

Description and Purpose

This modification upgraded the control system for MS-PCV-1139 in order to prevent overpressurization of the control system components, an event that can prevent the functioning of MS-PCV-1139. The scope of the modification entailed replacing the Solenoid Operated Valves (SOVs) and installing a relief valve in the control air tubing upstream of these SOVs.

Summary of Safety Evaluation

The replacement of the SOVs and the installation of a relief valve in the control system for MS-PCV-1139 was an upgrade to the existing system because the potential for overpressurization was eliminated. The differences in the SOV's parameters were evaluated and raised no concern. Parameters such as weight, watt rating, working pressure and temperature, seats and gaskets were the same; therefore, tubing supports, the DC electrical distribution system and QA Category remain unchanged., Thus, the functionality of MS-PCV-1139 as described in its Design Basis Document and the FSAR was unchanged. In conclusion, this modification improved the system's reliability and performance; therefore, this modification did not involve an unreviewed safety question.

1995 ANNUAL REPORT

CLAS 94-3-058 HPC, REV. 1

RECLASSIFY HOT PENETRATION FLEX CONNECTORS, BUTTERFLY VALVES, INLET FILTERS FOR BLOWER 33 & 34 AND SILENCERS

Description and Purpose

This document evaluated the reclassification of the Hot Penetration Cooling System (HPCS) flexible connectors, inlet filters for the blowers, silencers, butterfly valves, non-vibrating pressure relief valves from Category I to Category M and the discharge and suction pressure indicators and the return line drain butterfly valves from Category I to Non-Category I.

Summary of Safety Evaluation

This classification did not increase the probability of an occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. The concrete would not reach the temperature levels that would cause structural failure. In order to lose significant structural properties, concrete must be held continuously at 500°F to 600°F. The hottest penetrations are the main steam lines, which normally operate at a temperature of 507°F. The heat transfer properties of the Main Steam penetrations are such that 507°F steam results in a concrete temperature of less than 200°F. The results of a two dimensional transient heat transfer analysis indicate that in the unlikely event that all cooling air is lost to the main steam penetration, the surrounding concrete would reach a maximum temperature of 200°F in approximately 100 hours and 280°F in approximately 1000 hours.

1995 ANNUAL REPORT

NSE 94-3-073 CVCS, REV. 0
(MMP 93-3-442 CVCS, REV. 0)

EVALUATION OF THE MODIFIED CONFIGURATION OF CONTAINMENT ISOLATION VALVES CH-MOV-250 A THRU D

Description and Purpose

The purpose of this Nuclear Safety Evaluation (NSE) was to determine the safety significance of the operation of IP3 with the modified configuration of containment isolation valves CH-MOV-250 A thru D which was determined to be outside the licensed/design basis for the plant. The NSE provides assurance that the plant can be operated safely until the next refueling outage.

Summary of Safety Evaluation

Following completion of modification MMP-93-03-442-CVCS, which connected isolation valve seal water to the gland leak-off connection on valves CH-MOV-250 A thru D, this NSE allows the plant to start and operate until the next refueling outage. It was determined that although valves CH-MOV-250 A thru D are outside the licensed/design basis for the plant, assurance exists that the plant can be operated safely and the consequences of an accident can be safely mitigated. This conclusion was based upon basic design and operation of the valves, Isolation Valve Seal Water System acting as a second barrier to the stem packing, and the Probabilistic Risk Assessment frequency of an accident of this type occurring and the conservative post accident leakage/release limits which are within the regulatory guideline.

1995 ANNUAL REPORT

NSE 94-3-087 IVSWS, REV. 0

EVALUATION FOR NOT REQUIRING POST-LOCA WETTING
OF STEM PACKINGS ON OUTBOARD GLOBE TYPE CIV'S
ON PROCESS LINES EGRESSING CONTAINMENT

Description and Purpose

This Nuclear Safety Evaluation (NSE) evaluates the safety significance of not providing post-LOCA seal water injection from the Isolation Valve Seal Water (IVSWS) to the stem packing of the outboard globe type Containment Isolation Valves (CIVs) on process lines egressing containment. The NSE demonstrated that IVSWS is not required to wet the stem packing of the outboard CIVs with the exception being the Steam Generator Blowdown CIVs where both the inboard and out board valve packings are wetted by IVSWS.

Summary of Safety Evaluation

The absence of seal water wetting the stem packing of the outboard containment isolation globe valve in outgoing lines, the isolation system will still provide two barriers to the release of containment atmosphere to the evirons and the reliability of those barriers will not be affected. The functional capability of the Containment Isolation System will not be affected. This evaluation determined that there were no unreviewed safety questions.

1995 ANNUAL REPORT

CLS 94-3-095 CVCS, REV. 0

IVSW/CVCS INTERFACE CHECK VALVES CLASSIFICATION

Description and Purpose

The purpose of this classification was to classify four new check valves who's function is to eliminate pressurization of the IVSWS from the CVCS. The check valves were installed under mod MMP 93-3-442 CVCS.

Summary of Safety Evaluation

The addition of these four new check valves do not in any way change, create, or increase the probability of occurrence or consequences of an accident or malfunction of plant Structures or components important to safety previously evaluated in the FSAR.
However, this class did require changes in drawings described in the FSAR.

1995 ANNUAL REPORT

NSE 94-3-125 CVCS, REV. 2

ANALYSIS OF REACTIVITY HOLD-DOWN REQUIREMENTS

Description and Purpose

This Nuclear Safety Evaluation (NSE) clarified the functional capability of the Boric Acid Transfer Pumps, and justified editing the FSAR to reflect a boration rate of 132 ppm/hr between the Chemical and Volume Control System and the Reactor Coolant System.

Summary of Safety Evaluation

This NSE provided recommended revisions to the FSAR, which clarify the minimum CVCS boration capability as 132 ppm/hr. This minimum required boration rate has been shown to provide adequate post-shutdown reactivity control. It is unnecessarily restrictive to require a substantially higher boration rate of 75 gpm. The illustrative statements in the FSAR that imply such requirements therefore should be edited as shown in Attachment 1 to ensure that they are not misinterpreted as required levels of system performance and to make them more consistent with actual plant operations.

1995 ANNUAL REPORT

NSE 94-3-176, REV. 5

IP3 SITE ORGANIZATION CHANGES/FSAR UPDATE

Description and Purpose

This Nuclear Safety Evaluation (NSE) evaluated organizational changes made at Indian Point 3. The following title changes were made: Resident Manager to Site Executive Officer, Technical Services Manager to System Engineering Manager, Contract Services Manager to Construction Services Manager, Information Officer to Manager of Communications, Senior Reactor Operator to Control Room Supervisor, Shift Supervisor to Shift Manager, and the creation of the Planning and Scheduling Manager, the Operations Review Group Manager, the Licensing Manager, the Central Planning Manager, the Assistant Operations Manager, the Watch Engineer, the Field Support Supervisor and the Control Room Supervisor positions, the reinstatement of the Assistant to the Site Executive Officer, the deletion of the Office Manager, the reassignment of position responsibilities involving work control, outage management, emergency planning and computer services.

Summary of Safety Evaluation

The organizational changes are aimed at improving operations at the site and do not alter Indian Point 3's commitment to maintain a management structure that contributes to the safe operation and maintenance of the plant. All administrative and operational functions previously performed continue to be performed although possibly not by the same department, group or position.

1995 ANNUAL REPORT

MMP 94-3-186 SI. REV. 0

REMOVAL OF BIT BYPASS VALVES SI-1833A AND 1833B

Description and Purpose

This modification removed containment isolation valves SI-1833A and SI-1833B in the BIT bypass line. To preclude the need to add or to modify piping supports on the affected line, 3/44"-#270, piping was installed in place of the removed valves. Line #270 was capped at two locations downstream of the location of SI-1833B; one on the to-be-retired in place bypass line, while the other is on a short section connecting to the BIT outlet line, 4"-#16. Line #270 was also cut and capped at two locations on its upstream end, which is inside the Safety Injection (SI) Pump Room and adjacent to the BIT inlet line, 6"-#550.

The Isolation Valve Seal Water System (IVSWS) injection line to the piping interspace between the two removed valves was cut and capped. A root valve in the IVSWS injection line, IV-1404, as well as an adjacent check valve, IV-1504, was removed from this injection line. This modification resulted in a change to the Containment Isolation Valve list in the Technical Specifications.

Summary of Safety Evaluation

This bypass line was originally provided for periodic surveillance testing of SI Pump #33; this was necessary because the BIT contained a relatively high (11½% nominal, by weight) boric acid concentration which was required to be maintained at an elevated temperature (for solubility purposes) and to be continuously circulated (for uniformity purposes). Plant Modification 86-03-150 SIS provided for functional elimination of the BIT; it now contains boric acid of RWST quality and no longer requires heat-tracing and fluid circulation.

1995 ANNUAL REPORT

MMP 94-3-208 WGA, REV. 0

REPLACEMENT OF PRESSURE REGULATOR GA-PCV-499 IN PRT SAMPLING SYSTEM

Description and Purpose

The purpose of this modification was to remove Temporary Modification 93-02540-02, which removed pressure regulator GA-PCV-499 and replaced it with 3/8" tubing to facilitate Pressure Relief Tank (PRT) gas space sampling, and provide a new Veriflow HFR 900K series regulator suitable for the waste gas analyzer operation. A bypass line around the pressure regulator was provided to furnish the capability to sample the gas space in the PRT should the regulator malfunction.

Summary of Safety Evaluation

The new pressure regulator restored the PRT sampling line to its previous operational and functional capability. The regulator was determined suitable for its intended function. The bypass line around the regulator was provided to facilitate gas sampling in case of regulator malfunction. The regulator and tubing were seismically supported to assure no detrimental impact on plant operation during normal and accident conditions. Therefore, there was no impact on safety related systems, structures, or components.

1995 ANNUAL REPORT

MMP 94-3-210 SWS, REV. 0

REPLACEMENT OF SWN-109-1 & SWN-109-2

Description and Purpose

This modification replaced the existing Service Water System discharge isolation valves SWN-109-1 & SWN-109-2 downstream of the Central Control Room Air Conditioner (CCRAC) units with butterfly valves. Flanges were added on the piping to allow the installation of the new butterfly valves and a small section of pipe was replaced.

Summary of Safety Evaluation

The existing valves were 150#, Velan butt welded globe valves Model #B10-00074B-2TS. The replacement was a 150# Neles-Jamesbury lugged, wafer sphere butterfly valve Model 3"-815L-11-3600-MT with a manual hand lever operator installed between a pair of weld neck flanges. Due to the occasional throttling use of this valve, the substitution increases system availability, and the flexible seat design ensures good isolation for this location.

An analysis by UE&C, refer to letter IUP-10169, examined the new piping arrangement and found the stress levels and hanger loadings to be acceptable without further modifications.

1995 ANNUAL REPORT

MMP 94-3-228 CL, REV. 0

CHLORINATION SYSTEM PIPING

Description and Purpose

This modification was to repair the existing Chlorination System such that releases to the environment of sodium hypochlorite be prevented. This was accomplished by 1) replacing the existing fiberglass reinforced piping (FRP) with piping of a design, and materials of construction better suited for this application, 2) routing the new piping to minimize the potential for leakage, i.e., minimizing the number of flanged connections and 3) protecting the piping from damage that can result in excessive leakage, either within buildings or within a guard pipe. Additionally, the new piping minimizes piping runs and piping supports, to allow piping inspection access and to facilitate installation.

Summary of Safety Evaluation

This modification replaced chlorination system supply piping with materials designed and configured to reduce the potential for accidental release of sodium hypochlorite. The potential for a postulated Sodium Hypochlorite spill causing a Control Room habitability concern was addressed in MOD 87-03-133 CW and found not to qualify for consideration in Control Room habitability analysis in accordance with Regulatory Guide 1.78. A 15% sodium hypochlorite solution is not known to give off toxic vapor and gas liberation from solution is extremely slow. Additionally, since the vapor pressure of sodium hypochlorite is well below 10 torr; it does not qualify for consideration in CCR habitability analyses (Reference 23). This modification decreased the volume of chlorination system piping and does not in any way add to or subtract from the total available inventory of sodium hypochlorite.

1995 ANNUAL REPORT

MMP 94-3-334 CRHV, REV. 0

REMOVAL OF THE CENTRAL CONTROL ROOM STEAM HEATING COILS, PIPING AND ASSOCIATED COMPONENTS

Description and Purpose

This modification removed the steam coil from the #31 and #32 Central Control Room (CCR) A/C Units to facilitate easier cleaning of the cooling coils and removed all associated unused steam heating equipment and components. The scope of the mechanical portion of this modification included the removal of steam piping and components downstream of valve UH-79, condensate piping and components upstream of valve UH-80, steam coils in the CCR A/C Units, and associated condensate pumps. The scope of the electrical portion of this modification included sparing cables and retiring in place equipment associated with the condensate pumps 35A, 35B, MCC 36A cubicle 7FD, and MCC 36B cubicle 7FD.

Summary of Safety Evaluation

This modification provided the capability to increase the cooling capacity of the safety related CCR A/C #31 and #32 Units and allow easier cleaning and inspection of the cooling coils. This modification removed unused equipment (Auxiliary Steam Heating coils and related components) from the system and did not introduce any new failure modes that are different from failure modes postulated for the existing configuration. The testing performed per ENG-564 also verified that support systems (ie., 480V) are within design tolerances.

1995 ANNUAL REPORT

NSE 94-3-284, REV. 1

ORGANIZATION CHANGES IN NYPA MANAGEMENT STRUCTURE AND NUC GEN DEPARTMENT

Description and Purpose

This evaluation ensures that no unreviewed safety question exist in reorganizing the New York Power Authority Management and the Nuclear Generation Department. The overall management structure of the New York Power Authority was modified to replace departments with new business units which provide a greater management focus in specific functional areas. The management structure of the Nuclear Generation Department was changed to facilitate the transfer of some management positions and most technical support staff from headquarters office in White Plains to the two nuclear facilities.

Summary of Safety Evaluation

The changes are administrative in nature and do not involve plant equipment or operating conditions.

1995 ANNUAL REPORT

MOD 94-3-346, REV. 0

ROD CLUSTER CONTROL SYSTEM IMPROVEMENTS [GL-93-04]

Description and Purpose

This modification implemented a revised Westinghouse standard timing sequence for rod drive mechanisms [as detailed in Westinghouse Technical Bulletin NSD-TB-94-05-RO], suppress a potential source of back-EMF voltage spikes [as detailed in Westinghouse Technical Bulletin NSD-TB-93-03-RO], and improved the overall reliability of the Control Rod Drive System. These changes result from commitments made by letters IPN-93-109 & IPN-94-134 to provide resolution to concerns identified in NRC Generic Letter 93-04.

Summary of Safety Evaluation

This modification was performed to address NRC Generic Letter 93-04 and commitments made by letters IPN-93-109 & IPN-94-134, to adjust the timing on the Rod Cluster Control System decoder cards to prevent uncontrolled asymmetric rod withdrawals during demand for rod insertion. The acceptance testing demonstrated that the electronics work properly, units indicate demand position properly, rods move on demand, and the CRDM current orders are not corrupted. Precautions indicated ensured core reactivity changes were not made during testing.

1995 ANNUAL REPORT

MMP 94-3-349 480V, REV. 0

EDG XFMR, BREAKER MCC-36B/6RBR FUSE REPLACEMENT

Description and Purpose

This modification corrected a discrepancy associated with the power fuses installed in 480VAC MCC36B cubicle 6RBR. The fuses installed in this cubicle were too large and did not provide proper protection and coordination of the electrical circuits. The scope of this modification was limited to the replacement of the existing power fuses in MCC 36B, cubicle 6RBR, with Buss LPJ-40SP fuses.

Summary of Safety Evaluation

The installation of this modification did not adversely affect any safety related systems or components needed for the safe shutdown and mitigation of an accident. Proper coordination of the fuses with the other electrical protection devices will prevent the inadvertent loss of additional equipment if a short circuit were to occur and will aid in maintaining the availability of safety related equipment and Appendix R power supplies.

1995 ANNUAL REPORT

MMP 94-3-372 IA, REV. 0

PROVIDE IA QUALITY SUPPLY SOURCE
TO 31 AND 32 IA COMPRESSOR UNLOADER VALVES

Description and Purpose

The purpose of this modification was to eliminate the common line interaction between the pressure switches and the unloader solenoid/compressor unloader valves. This modification provides a separate instrument air quality supply source to the IA compressor unloader SOV's. The new air supply, down stream of the desiccant dryers, alleviates concerns arising from the wet air supply. As a result of this modification, the problems with the unloader SOV's will be eliminated.

Summary of Safety Evaluation

The new supply line is functionally equivalent to the original and will provide moisture free Instrument Air to the unloader SOV's and was determined to have no safety consequences but required a change to a drawing described in the FSAR.

1995 ANNUAL REPORT

NSE 94-3-374 RM, REV. 0

EVALUATION FOR THE IMPLEMENTATION OF A NEW METHOD FOR OBTAINING BACKUP HIGH RANGE PLANT VENT READINGS DURING OFF-NORMAL CONDITIONS

Description and Purpose

This Nuclear Safety Evaluation (NSE) implemented a new method for obtaining backup high range plant vent radiation monitoring during off-normal conditions in the event that R-27, Wide Range Plant Vent Monitor, fails. This new method obtains high range radiation readings off of the plant vent sample lines in the post accident plant vent sampling system shield. Revise FSAR to remove the references to the current methodology.

Summary of Safety Evaluation

The new method for obtaining backup high range plant vent radiation readings does not result in changes to Technical Specifications, or impact on any safety related or environmentally qualified structures, systems or components. The upgrade to the new method will require appropriate revisions of the FSAR to reflect the changes evaluated herein. Instituting the new methodology for obtaining backup high range plant vent radiation readings is acceptable and does not involve an unreviewed safety question.

1995 ANNUAL REPORT

NSE 94-3-380 ED, REV. 2

EOP'S REVISION IMPACT TO SAFEGUARDS BUS LOADING

Description and Purpose

This Nuclear Safety Evaluation (NSE) was performed to ensure that the latest revisions to the Emergency Operating Procedures (EOP) will not result in the 480V Safeguards Switchgears (2A,3A,5A,and 6A) to exceed their design margins for load carrying capacity and that the EOPs will not overload the Emergency Diesel Generators (EDGs) by exceeding the EDG continuous ratings of 1750 Kw for more than two (2) hours and a maximum peak rating of 1950 Kw.

Summary of Safety Evaluation

This evaluation determined that the 480V Buses, supply breakers and station service transformers can function properly for peak loadings of 3600 Amps for a sufficient length of time to handle peak accident loading periods. Additionally the buses, supply breakers and transformers can withstand currents of up to 4400 Amps for a limited amount of time. However, the load management presently in effect ensures that the peak bus loading does not exceed 3600 Amps under any circumstances, and are below 3200 Amps at the end of all scenarios after operator/ procedure actions are completed.

1995 ANNUAL REPORT

NSE 94-3-394 N2, REV. 0

PORV NITROGEN (N_2) SUPPLY FROM SI ACCUMULATORS

Description and Purpose

The Nuclear Safety Evaluation (NSE) addressed providing a continuous N2 source for the pressurizer power operated relief valves from the safety injection accumulators.

Summary of Safety Evaluation

Since the two PORV accumulators, at six cubic feet of volume were connected to the SI Accumulators at 1,100 cubic feet, the overpressurization protection system was able to exceed its required design capability and ensure necessary operation potential. Temporary Modifications 94-04169-08, 95-02751-05, 95-04473-00, and 95-04473-03 documented the connection for each use.

The load on the N2 system does not cause reduction in the capability of charging the SI accummulators and does not present system interaction concerns.

1995 ANNUAL REPORT

CLS 94-3-448 IAS, REV. 1

INSTRUMENT AIR TUBING BETWEEN ROOT VALVES AND THE AIR OPERATORS

Description and Purpose

The purpose of this reclassification was to change the Q.A. Category of specifically delineated portions of instrument air (IA) tubing from Category I to Category M and to reclassify the affected portions of the IA tubing from Seismic Class I to Seismic Class III. In addition the Q.A. Category of a portion of the nitrogen supply line to, and the IA vent line from Atmospheric Relief Valves MS-PCV-1134, -1135, -1136, and -1137 from Category I to Category M and the seismic class from Class I to Class III.

Summary of Safety Evaluation

This reclassification does not increase the probability of occurrence or the consequences of an accident or malfunction of structures, systems or components important to safety previously evaluated in the FSAR. The change in classification does not affect the operation or the performance of the IA System and therefore does not prevent the safety-related valves or dampers from failing to their safe position.

NSE 95-3-011 PABHV, REV. 0

PAB HEPA FILTERS FRAME CONSTRUCTION MATERIAL

Description and Purpose

This Nuclear Safety Evaluation (NSE) provides justification for allowing the use of either stainless steel type 409 or type 304L, for the construction of HEPA filter frames (previously 304L was the only approved material).

Summary of Safety Evaluation

Per Reg. Guide 1.52, ANSI N509 and MIL-F-51068 both type 409 and 304L stainless steel are acceptable material for the construction of HEPA filter frames for all nuclear filtration systems at IP-3.

1995 ANNUAL REPORT

NSE 95-3-032, REV. 1

EVALUATION OF IP3 REGULATORY GUIDE 1.97 INSTRUMENTATION AND CABINET TEMPERATURE RISE

Description and Purpose

The purpose of this Nuclear Safety Evaluation (NSE) was to demonstrate that the minimum set of required post-accident instrumentation in the IP3 Central Control Room (CCR) will remain operable when exposed to the elevated temperatures (CCR ambient temperature plus cabinet temperature rise) resulting from a post LOCA scenario with one safety-related air conditioning (A/C) train inoperable.

Summary of Safety Evaluation

This NSE demonstrated reasonable assurance that the minimum set of required post-accident instrumentation in the IP3 Central Control Room (CCR) will remain operable when exposed to the elevated temperatures (CCR ambient temperature plus cabinet temperature rise) resulting from a post LOCA scenario with one safety-related air conditioning (A/C) train inoperable.

The above described accident scenario will result in a long term steady state ambient bulk air temperature of the CCR of 106°F. The instrumentation found in enclosed cabinets, panels, or racks would also experience additional temperature rise across the enclosure which must be accounted for in evaluating their continued operability.

1995 ANNUAL REPORT

MMP 95-3-033 MS/FW/HD, REV. 0

REMOVE CCR RECORDERS TR-1104, TR-1108, FR-1106, FR-1119

Description and Purpose

This Nuclear Safety Evaluation (NSE) evaluated the acceptability of removing four recorders from the Central Control Room (CCR) Supervisory Panel, and the retiring of instruments providing the signals to these recorders. The recorders and the parameters they monitor are: TR-1104 Pen 1 Boiler Feed Pump Discharge Temperature, and Pen 2 High Pressure Heater Discharge Temperature; TR-1108 Pen 1 Reheater Drain Tank Temperature 3A, and Pen 2 Reheater Drain Tank Temperature 3B; FR-1106 Pen 1 Reheater Drain Tank Flow 3A, and Pen 2 Reheater Drain Tank Flow 3B; FR-1119 Pen 1 Reheat Steam to Boiler Feed Pump 31, and Pen 2 Reheat Steam to Boiler Feed Pump 32.

Summary of Safety Evaluation

This activity is removing CCR recorders TR-1104, TR-1108, TR-1106, and FR-1119. These recorders monitor Boiler Feed Pump and High Pressure Heater discharge temperatures, Reheater Drain Tank Drain temperatures for Tanks 3A and 3B, Reheater Drain Tank Drain Flows for Tanks 3A and 3B, and Reheat Steam to Boiler Feed Pump 31 and 32 turbine respectively. These recorders do not provide any information that is utilized to mitigate the consequences of any accident evaluated in the SAR. Additionally, the instruments that provide the signal to these recorders will be retired (pending future use) since they do not provide any other function with any other system except to drive the four (4) recorders being removed.

1995 ANNUAL REPORT

NSE 95-3-044 PZR, REV. 0, 1, 2

OPERATION WITH A STEAM BUBBLE IN THE PRESSURIZER AND THE RCS AT COLD SHUTDOWN

Description and Purpose

This Nuclear Safety Evaluation (NSE) evaluated the operating conditions resulting from creation of a steam bubble in the pressurizer when reactor coolant bulk temperature is below 200°F and containment integrity is not established.

Summary of Safety Evaluation

A typical plant startup includes starting a single Reactor Coolant Pump (RCP) when the Reactor Coolant System (RCS) is pressurized to above 325 psig, with T_{avg} below 200°F and the system solid (i.e., pressurizer is full). If the RCP is to be used to heat the RCS above cold shutdown, containment integrity is established prior to pump start. Once pump heat brings T_{avg} above 200°F, then a steam bubble is drawn in the pressurizer, and the heatup process continues.

This process includes the potential of a pressure spike in the RCS resulting from pump start. If there were a steam bubble in the pressurizer, the pressure spike resulting from subsequent pump starts would be almost completely absorbed by the gas cushion, with little or no observable effect on RCS pressure. When the RCS is solid, with no bubble in the pressurizer, there is no gas cushion to absorb this small jump in pressure, and there may be a challenge to the Overpressurization Protection System (OPS), which would automatically depressurize the RCS through the pressurizer Power Operated Relief Valves (PORVs).

Operation with a steam bubble in the pressurizer, the rest of the RCS at cold shutdown, and conatinment integrity relaxed was shown in this evaluation to be consistent with the safety analyses as described in the FSAR. Any postulated accident involving loss of RCS inventory under these conditions, either via the pressurizer or anywhere else in the RCS, is bounded by analyzed scenario.

1995 ANNUAL REPORT

MMP 95-3-056 CBHV, REV. 0

REMOVAL OF CO_2 INTERLOCK 15' CONTROL BUILDING FANS

Description and Purpose

This minor modification (MMP) removed the interlock between the CO_2 Fire Protection System and Switchgear Room Exhaust Fans 33 and 34 to preclude potential failure of the room's ventilation system due to the failure of a single component associated with the interlock. This interlock was replaced with contacts from the new manual switches in the control circuits for Exhaust Fans 33 and 34, Louver L-319 and Fire Dampers FD-1, FD-2 and FD-9.

Summary of Safety Evaluation

Minor modification MMP 95-3-056 CCBHV was issued to resolve a single failure concern in the 15 foot Control Building, Switchgear Room Ventilation System. A postulated hot short on the CO_2 interlock relay in the control circuit of CB Switchgear fans 33 and 34 and dampers FD-1, FD-9 and louver L-319 could have resulted in the loss of all ventilation. The loss of the Switchgear Room ventilation could affect the operability of the 480V vital switchgear under certain operating conditions. The proposed modification removed the CO_2 system interlock from the control circuits of the Switchgear Room ventilation fans and dampers and louver so that the ventilation system would not be inoperable on a single failure.

Two new manually controlled switches were added to provide independence and separation of the fan controls. The modified system requires operator action to manually initiate the CO2 system and close the louvers and vents and stop the ventilation fans after a fire has been verified.

1995 ANNUAL REPORT

NSE 95-03-067 CCW, REV.1

SAFETY INJECTION CIRCULATING WATER PUMPS REPLACEMENT

Description and Purpose

The purpose of this Nuclear Safety Evaluation (NSE) was to determine the safety significance of adding a 3/4 inch throttling valve in the lines to each of the two seal jacket coolers associated with each of the three High Head Safety Injection (HHSI) Pumps and the inservice test connections for each of the three Safety Injection Circulating Water Pumps. The throttling valves were added to redistribute the flow to HHSI pump coolers to ensure minimum required CCW flow for the lube oil cooler.

Summary of Safety Evaluation

The addition of throttling valves in the lines to the seal jacket coolers associated with the HHSI pumps does not affect the functional capability of the Safety Injection System. The new valves improve the flow distribution to ensure that the HHSI pump lube oil cooler is adequately cooled. Therefore, the probability of occurrence of an accident or malfunction previously evaluated in the FSAR was not increased. The modification improves the flow to the most critical component, the lube oil cooler.

1995 ANNUAL REPORT

NSE 95-03-102 FP, REV. 0

FIRE PROTECTION SYSTEM PIPING SUPPORTS SEISMIC CLASS

Description and Purpose

The purpose of this Nuclear Safety Evaluation (NSE) was to evaluate any safety concerns associated with a change to the FSAR chapter 9.6, Facility Service System and chapter 16.1, Seismic Design Criteria for Structures and Equipment. The change involved the seismic class of the fire protection piping supports in Category I areas of the plant.

Summary of Safety Evaluation

The changes to fire protection piping supports seismic design criteria does not change the function, performance, or integrity of any boundaries with which safety related systems form or support the primary protective barriers on which the consequences of previously identified accidents are based. Therefore, the consequences of previously identified accidents remain bounded by the results contained in the FSAR.

1995 ANNUAL REPORT

NSE 95-03-104 IAS, REV.1

INSTRUMENT AIR TUBING RECLASSIFICATION

Description and Purpose

The purpose of this Nuclear Safety Evaluation (NSE) was to determine the safety significance of reclassification CLAS 94-03-448 IAS which changed the Q.A. Category of specifically delineated portions of instrument air (IA) tubing from Category I to Category M, and affected portions of the IA tubing from Seismic Class I to Seismic Class III. In addition, the classification of a portion of the nitrogen supply line to, and the vent line from, the atmospheric relief valves MS-PCV-1134, -1135, -1136, and -1137 was changed in the same manner. This NSE documents the changing clasification.

Summary of Safety Evaluation

The portions of the IA System tubing that were reclassified were never relied upon for causing or allowing the air operated valves or dampers to change to their fail-safe positions. Therefore, the reclassification has no impact on the ability of those valves and dampers to perform their safety function. In the case of the atmospheric relief valves a small portion of nitrogen tubing was reclassified, however isolation from the Q.A. CategoryI/Seismic Class I portion of the nitrogen tubing (required for the Aux. Feed Water System) is accomplished by manual valves. To assure the system will perform its isolation function, the first support in the newly reclassified portion of tubing was maintained Seismic Class I.

1995 ANNUAL REPORT

NSE 95-3-114 SGS, REV. 0

SGS CHLORIDE ANALYZER CL-1101

Description and Purpose

This Nuclear Safety Evaluation (NSE) was written to support Temporary Modification (TM) No. 94-05698-03. This TM was in response to DER 94-1334, written by the Radiological Environmental Services Department (Plant Chemistry). As documented in DER 94-1334, the original steam generator (SG) blowdown chloride analyzer was replaced with an ion chromatograph prior to 1992 without proper authorization or documentation.

Summary of Safety Evaluation

The Ion Chromatograph Analyzer is part of the Steam Blowdown Sampling System. This system is Non Category I. It is not used to safely shutdown the unit, prevent or mitigate a release, or to mitigate core damage. In addition, it does not support the operation of safety related equipment. Its intended function is to measure impurities in the secondary steam generator water which may cause steam generator degradation. By measuring these impurities, and subsequently adjusting the feedwater chemistry, steam generator corrosion may be thwarted.

Having the ion chromatograph on-line has several benefits. Not only does it minimize personnel error in sampling techniques, but it enhances analysis sensitivity. Steam generator chloride and sulfate concentrations are typically below 2 PPB. This level of concentration cannot be accurately measured by analyzing grab samples due to contaminants introduced by the environment. However, an on-line system can accurately detect such minuscule concentrations, facilitating chemistry changes which will extend the life of the steam generators.

1995 ANNUAL REPORT

NSE 95-3-135, REV. 0

EVALUATION OF HIGH HEAD SI SYSTEM FLOW BALANCE AND PUMP PERFORMANCE

Description and Purpose

This Nuclear Safety Evaluation (NSE) addresses recent activities which impact performance characteristics of the High Head Safety Injection (HHSI) System. The first activity relates to the HHSI flow balance as controlled by the positions of the branch line throttle valves (SI-856A through K). The second activity involves the refurbishment of all three Safety Injection Pumps (SIPs), which had the potential to effect changes in the performance characteristics of these pumps. The branch line flow balance and SIP capability are the two most critical variables in the HHSI flow calculations upon which the safety analyses are based. As such, this safety evaluation addresses the effect of these activities on the ability of the HHSI System to perform in accordance with the safety analyses.

Summary of Safety Evaluation

This safety evaluation addresses the existing HHSI system flow balance and pump performance with regards to the ability of the system to operate within the limits of the safety analyses and the design bases. The accuracy of instruments used to set the branch line flow balance in 1992 had been questioned, and required evaluation. Also, all three Safety Injection (SI) pumps had been refurbished, with new rotating elements installed on the 31 and 33 SI pumps. A series of tests collected data on the as-found system flow capability and on the performance of each pump. The results were evaluated by Westinghouse, who issued two reports concluding that the hydraulic performance of the HHSI system conforms with all safety analysis and design basis requirements. Based on the results of these Westinghouse reports, combined with evaluations of the SI pump motor KVA limits and measurement accuracy during the various tests, it is concluded that no unreviewed safety question exists.

1995 ANNUAL REPORT

NSE 95-03-185 WCCPP, REV. 0

SCR FOR WCCPPS N2 PRESSURE SWITCHES

Description and Purpose

The purpose of this evaluation was to support Setpoint Change Request (SCR) IP3-94-0021, which changed the setpoint of Weld Channel & Containment Penetration Pressurizaion System (WCCPPS) nitrogen pressure switches PC-1201S through 1204S from 43 psig to 45 psig.

Summary of Safety Evaluation

The change did not create a new failure mechanism or change the manner in which any system, structure, or component performs its' intended function. The setpoint change facilitates IP3's compliance with Technical Specification 3.3.D.1.a by ensuring nitrogen is supplied to a WCCPPS zone prior to the zone's pressure dropping below 43 psig. It was therefore concluded that the setpoint change did not present an unreviewed safety concern.

1995 ANNUAL REPORT

NSE 95-3-187 SWS, REV. 0

EVALUATION OF POST-LOCA (EXTERNAL RECIRCULATION) INACCESSIBILITY OF SERVICE WATER CONTAINMENT ISOLATION VALVES (SWN-51-1 THROUGH-5, SWN-71 THROUGH-5)

Description and Purpose

This NSE evaluated the inaccessibility of service water containment isolation valves (Fan Cooler Unit valves SWN-51-1 through SWN-51-5 and Fan Cooler Unit motor cooler valves SWN-71-1 through SWN-71-5) during post-LOCA external recirculation and deleted the implication (from sections 6.4, 6.7.1, 9.6.1 and 11.2.3.1 in the FSAR) that the capability to identify and isolate a leaking FCU/FCU motor cooler is required for all post accident conditions.

Summary of Safety Evaluation

The deletion of the capability to identify and isolate a failed FCU or FCU motor cooler during external recirculation does not increase the possibility of any accidents previously evaluated in the safety analysis report because the need to perform this action is only required if a coil failure is concurrent with the Design Basis Accident, which is already considered in the safety analysis report.

1995 ANNUAL REPORT

NSE 95-03-258 PZR, REV. 0

TECHNICAL BASIS FOR IMPLEMENTATION OF TIME LIMITS IN PROCEDURE ARP-3

Description and Purpose

The purpose of this Nuclear Safety Evaluation (NSE) was to provide a technical basis to support changes to Alarm Response Procedure ARP-3 under Term Procedure Change TPC-95-1051.

Summary of Safety Evaluation

Term Procedure Change TPC-95-1051 was written to disallow intentional reduction of RCS pressure below 2205 psig in response to "Pressurizer PORV and Safety Acoustic Monitoring" and "Pressurizer Safety Valve Outlet High Temp" alarms. This NSE confirmed that there is sufficient technical support for the Limiting Condition for Operation (LCO) requirements presented in TPC 95-1051. The implementation of the TPC (or subsequent revision of procedure) does not increase the probability of occurrence of an accident evaluated in the FSAR. The likelihood of a design basis accident neither increases nor decreases during low pressure operation while recovering from this LCO.

1995 ANNUAL REPORT

NSE 95-03-302 RCS, Rev. 0

REACTOR VESSEL FLANGE "O" RING REPLACEMENT

Description and Purpose

The purpose of this Nuclear Safety Evaluation (NSE) was to demonstrate that Type 1 Change DC 95-03-203 RCS which installed new metallic "O"-rings provided by the Helicoflex Company to seal the reactor vessel head to the top flange of the reactor vessel.

Summary of Safety Evaluation

The new "O"-rings have a slightly larger cross-section which improves the sealing margin and provides better sealing over irregularities in the seal seating surfaces. The change does not challenge the design limits of any RCS components and does not impact any systems subject to operational transients or design basis accidents. There is no change to the mechanical operation of the reactor vessel or any other RCS component and the primary pressure boundary will be maintained.

1995 ANNUAL REPORT

NSE 95-03-304 SI, REV. 0

EVALUATION OF CHANGES TO EMERGENCY OPERATING PROCEDURES MADE DURING THE RCIP/RS94 OUTAGE

Description and Purpose

The purpose of this Nuclear Safety Evaluation (NSE) was to justify changes made to the Emergency Operating Procedures during the RCIP/RS94 outage.

Summary of Safety Evaluation

The NSE justified changes to the EOPs to incorporate enhancements to EOP setpoint calculations, plant modifications, and engineering evaluations of procedural enhancements to improve usability and maintain consistency with the assumptions made in the plants' safety analysis. The analysis and engineering evaluations show that all changes increase or maintain an adequate margin to safety and improve the plants' response to accidents within and beyond the design basis.

1995 ANNUAL REPORT

NSE 95-3-305 EDG, REV. 0

31 EDG ISOLATION SWITCHES

Description and Purpose

This NSE provided the basis for removing the provision for 31 EDG isolation from the FSAR. Isolation switch No's 1 through 10, (located in the Switchgear Isolation Cabinet, elevation 15 in the Control Building) which supplied power to the required safe shutdown equipment have been removed and alternate power is supplied from the Appendix "R" Diesel Generator.

Summary of Safety Evaluation

The 31 EDG isolation switches are no longer relied upon for Appendix "R" compliance for a fire in the Cable Spreading Room or Control Room. Power for the safe shutdown equipment required during a postulated Appendix "R" fire is supplied by the Appendix "R" Diesel Generator.

1995 ANNUAL REPORT

NSE 95-3-309 COND, REV. 1

EVALUATION OF REVISION TO FSAR FIGURE 10.2-3B

Description and Purpose

This NSE revision corrected a typographical error concerning the seismic classification of the Turbine Building and identifies an additional drawing 9321-F-70203 to the list of drawings requiring revision.

Summary of Safety Evaluation

The changes made in this revision were typographical and have no safety significance.

1995 ANNUAL REPORT

NSE 95-3-310 FP, REV. 0

REMOVAL OF CONTROL ROOM HVAC DUCT SMOKE DETECTORS

Description and Purpose

The purpose of this nuclear safety evaluation (NSE) was to evaluate the safety significance associated with the elimination of the duct mounted smoke detectors installed in the Control Room HVAC return ductwork.

Summary of Safety Evaluation

The Control Room has area wide smoke detection installed at ceiling level and in cabinets throughout the Control Room. The removal of the duct mounted smoke detectors does not affect the ability of the area wide detection system, is consistent with the National Fire Codes and fully meets the intent of the NRC staff position on detection.

1995 ANNUAL REPORT

NSE 95-3-314 VCHVP, REV. 0

CONTAINMENT PURGE SUPPLY AND EXHAUST VALVES STROKE TIME

Description and Purpose

FSAR Section 5.1.4.2.2 describes the operations of the containment supply and exhaust purge ventilation system. The tight-sealing valves (one inside and one outside of the containment) are used for isolation purposes. The valves are manually opened for containment purging, but are automatically closed upon a signal of high containment pressure or high containment radiation level.

FSAR Section 5.1.4.2.2 will be revised to state that if for any reason, any of the four valves fail to open within a given time after the cycle is initiated, all four valves will close and pressure will be restored. The FSAR presently reads that if the valves fail to open within a given time (approximately 35 seconds) after the cycle is initiated, the valves will close and pressure will be restored.

Summary of Safety Evaluation

This activity will not increase the probability of a malfunction of any equipment important to safety previously evaluated in the SAR. No aspect of this activity poses a credible event that affects the "Safety Function" of the VC purge ventilation system. The purge valves are closed during reactor power operations and function as VC isolation valves during accident conditions requiring VC isolation.

1995 ANNUAL REPORT

NSE 95-3-316 WCCPPS, REV. 0

PENETRATION TEST FOR WATER LEAKAGE

Description and Purpose

This nuclear safety evaluation (NSE) was written to support revision 1 of OPerations Department Procedure 3PT-W16, Penetration Test for Water Leakage, to verify containment penetration SS was free of water leakage. The procedure was considered a test not anticipated in the FSAR which could temporarily degrade a portion of the Weld Channel and Containment Penetration System (WCCPPS) barrier outside the vapor containment building, thus requiring demonstration that no unreviewed safety question existed.

Summary of Safety Evaluation

Performance of this test cracks open WCCPPS isolation valves associated with eleven penetrations flow through test connections and the sample valves associated with three supply racks for 10 seconds to verify water is not present inside the penetration.

The containment penetrations have two barriers to preclude containment atmosphere from leaking through the penetration opening. In the event the flow through test connection or sample valves failed open during the performance of the test, the barrier outside the VC would be breached. However, the end plate on the VC side of the penetration would still be in place to effectively prevent containment leakage. As such, a release pathway for containment atmosphere to escape to the environment would not be created.

1995 ANNUAL REPORT

NSE 95-3-318 MULT, REV. 0

REVISION TO ADMINISTRATIVE PROCEDURE LISTINGS IN FSAR

Description and Purpose

This NSE justifies combining the programmatic and responsibility sections of AP-24.2 (Hazardous and Industrial Waste Management) and AP-46 (Chemical Material Control Program) into one procedure (AP-61, Chemical Material Control and Waste Management). The specific implementing steps were moved to a lower tier station directive (RES-SD-01) to be in accordance with AP-3.

Summary of Safety Evaluation

The changes made are administrative and do not affect the overall program or procedural requirements of Chemical Material Control or Waste Management.

1995 ANNUAL REPORT

NSE 95-03-323 EDG, REV. 0

EDG FUEL OIL DAY TANK MINIMUM LEVEL REQUIREMENT

Description and Purpose

The purpose of this Nuclear Safety Evaluation (NSE) was to evaluate the existing licensing basis for the Emergency Diesel Generator (EDG) fuel oil day tank level requirements and to define a new licensing basis for the low level setpoint which is currently 65%. This was in support of DER 95-2387.

Summary of Safety Evaluation

The NSE concluded that there is no licensing basis or technical need for a 55 minute, full load supply of fuel in the day tank. No fuel is required in the day tank to maintain a static head pressure on the injection manifolds or the booster pump, and there is no licensing basis or technical requirement for a 75 minute supply of fuel in the day tank post-accident. The existing system has a "start fuel transfer" at the 65% level, and a low level alarm at 30%.

1995 ANNUAL REPORT

NSE 95-03-324 VC, REV. 0

REVISE SECTION 11.2 OF THE FSAR REGARDING VC PRESSURE

Description and Purpose

The purpose of this Nuclear Safety Evaluation (NSE) was to evaluate revision of the FSAR Section 11.2.3.1, "Process Radiation Monitoring System." The following statement on FSAR page 11.2-11 Rev.11 was misleading and inconsistent with other statements in the FSAR and Technical Specifications: "The containment ventilation system provides means to manually limit, under normal conditions, containment pressure to 1 psig."

On the contrary, IP3s design and licensing basis documents consistently specify an acceptable range for containment pressure between 2 psig vacuum to 2.5 psig pressure, which bounds that specified in SOP-CB-3, Containment Pressure Relief and Purge Systems Operation. Consequently, FSAR section 11.2.3.1 should be revised to clearly state 1 psig is not an operating limit, consistent with the rest of IP3's design and licensing basis.

Summary of Safety Evaluation

The FSAR change does not impact the manner in which the unit is operated, it reconciles a documentation inconsistency. In addition, it does not degrade the margin of safety during normal operations or anticipated transients or the adequacy of systems, structures, or components to prevent accidents or mitigate accident conditions.

1995 ANNUAL REPORT

NSE 95-3-342 H2, REV. 0

MAIN GENERATOR GAS SYSTEMS OPERATION

Description and Purpose

As a result of an assessment of Generic Letter 93-06, this NSE reviewed the change in the position of valves HS-29, HS-39 and HS-494 from normally open to normally closed. These valves originally maintained pressure in the Main Generator. Pressure is now maintained by batch addition and the valves will only be used for makeup H2.

Summary of Safety Evaluation

The changes made are an enhancement and do not affect the function of the hydrogen system. In addition the operation of the hydrogen system valves are not safety significant and its function is not included in the safety analysis report.

1995 ANNUAL REPORT

NSE 95-03-352 RCC, REV. 0

OPERATOR RESPONSE TO INDICATION OF MISALIGNED RCCA

Description and Purpose

The purpose of this Nuclear Safety Evaluation (NSE) was to clarify the actions required in response to the indicated misalignment of a Rod Cluster Control Assembly (RCCA).

Summary of Safety Evaluation

All actions taken in response to a potential misaligned RCCA are consistent with the requirements of the Technical Specifications. Therefore, the procedural changes identified in the NSE do not increase the probability of occurrence of an accident evaluated in the FSAR. The slight Rod Position Indication (RPI) drift resulting from thermal imbalance within the RPI system is an observed phenomenon and has been documented as such in the industry. Allowing the RPIs to stabilize is an appropriate and conservative action. By presuming the affected RCCAs to be truly misaligned until the alarm clears or RCCA position is determined by alternate means, the operators are taking a conservative and proactive approach to the situation. Therefore, the procedural changes identified in this NSE do not involve an unreviewed safety question.

1995 ANNUAL REPORT

NSE 95-3-353 RCS, REV. 0

pH AND LITHIUM CONTROL IN THE REACTOR COOLANT SYSTEM

Description and Purpose

In early operating cycles at IP3, the peak boron concentration in the RCS for full-power equilibrium operation never exceeded 1200 ppm. With the advent of longer fuel cycles, the peak RCS boron concentrations have risen as high as 1300 ppm during power production (for Cycle 8), with higher concentrations required for maintenance of shutdown margin with the reactor subcritical. These higher boron concentrations have required lithium concentrations in the RCS (added as LiOH) greater than the original limit of 2.2 ppm in order to maintain a pH of 6.9 or greater in the RCS.

Because the original FSAR dealt with lithium concentrations no higher than 2.2 ppm, this NSE presents the evaluations performed to date to justify past and future plant operation with a lithium concentration of up to 3.5 ppm in the RCS.

Summary of Safety Evaluation

By controlling RCS chemistry under the strategy described in this NSE, fuel clad corrosion and crud deposition effects will be minimized, with no adverse effects on the tritium production rate. Therefore, this revised program for boron/lithium control does not increase the probability of occurrence of an accident evaluated in the safety analysis report.

This strategy for RCS chemistry control assures the best conditions for fuel cladding, which minimize the radiological consequences of any design basis accident or transient. Therefore, this revised program for boron/lithium control does not increase the consequences of an accident evaluated previously in the safety analysis report.

1995 ANNUAL REPORT

NSE 96-3-003, REV. 0

IP3 SITE ORGANIZATION CHANGES

Description and Purpose

The purpose of this nuclear safety evaluation was to describe and evaluate organization changes at Indian Point 3 which include the creation and deletion of positions and the reassignment of position responsibilities and reporting relationships. The following changes were made: The position of General Manager Operations (GM-OPS) was reestablished with Operations, Site Planning and Outage Services, Radiological and Environmental Services, Performance and Reliability, and Training reporting to the GM-OPS. The position of General Manager Support Services was eliminated. Emergency Planning, Security, and some materials management activities now report to the Site Executive Officer. The Computer Services Manager now receives direction from the General Manager Maintenance and continues to report to the Director Information Services in the headquarters office. The Operations Review Group now reports to the Tactical Assessment Coordinator (TAC). The Licensing Manager now reports to the TAC. The TAC now reports to the Director, Regulatory Affairs and Special Projects who now reports to the Vice President Appraisal and Compliance Services. In addition the title Vice President Regulatory Affairs and Special Projects was changed to Director Regulatory Affairs and Special Projects.

Summary of Safety Evaluation

The changes do not alter the Power Authority's commitment to maintain a management structure that contributes to the safe operation and maintenance of the plant.

1995 ANNUAL REPORT

TEMPORARY MODIFICATION TM 95-02751-01

RX O-RING LEAK MONITORING

Description and Purpose

The purpose of this temporary modification (TM) was to provide a connection and pressure gauge upstream of RC-AOV-544 to allow measurement of leakage past reactor vessel inner and outer O-rings.

Summary of Safety Evaluation

The TM installed a drain tap to allow quantification of leakage and affected FSAR figure 4.2-2A which depicts the line in which the modification was made. The TM did not create the probability of occurrence or create the possibility of an accident or malfunction of a different type than any other previously evaluated in the FSAR.

1995 ANNUAL REPORT

TEMPORARY MODIFICATION TM 95-03747-01

RCDT REGULATOR BYPASS

Description and Purpose

The purpose of this Temporary Modification (TM) was to bypass failed pressure regulator WD-PCV-1069D which regulates reactor coolant drain tank (RCDT) gas sample pressure to the waste gas analyzer. The regulator was removed and replaced by a length of tubing and associated fittings.

Summary of Safety Evaluation

To prevent over-pressurizing the Waste Gas Analyzer System, RCDT pressure must be less than 8 psig when sampling. This is ensured via connection to the vent header when sampling. Sampling is done manually in accordance with the appropriate procedures. FSAR figures 11.1-2C and 11.1-1A are temporarily affected by the TM. No permanent change to these figures was introduced by the TM. The overall function of the RCDT sampling system as described in the FSAR is unchanged.