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United States Nuclear Regulatory Commission
Mail Station P1-137
Washington, D.C. 20555

Attention: Document Control Desk

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT
DOCKET NO. 50-333
ANNUAL SUMMARY OF CHANGES, TESTS, AND
EXPERIMENTS FOR 1992

Enclosure: 1) Annual Summary of JAFNPP Changes, Tests, and Experiments for 1992

Gentlemen:

Enclosed is a summary of the changes, tests and experiments implemented at the James A. FitzPatrick Nuclear Power Plant during 1992.

This report provides the Nuclear Safety Evaluation number (e.g. JAF-SE-92-001) followed by a brief description of the corresponding change, test, or experiment and safety evaluation summary as required by 10CFR50.59(b) (2).

Very truly yours,

HARRY SALMON Jr.

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Introduction to the 1992 Annual 10CFR50.59 Report

10CFR50.59 states:

(a) (1) The holder of a license... may (i) make changes in the facility as described in the safety analysis report, (ii) make changes to the procedures as described in the safety analysis report, and (iii) conduct tests or experiments not described in the safety analysis report, without prior Commission approval, unless the proposed change, test or experiment involves a change in the technical specifications incorporated in the license or an unreviewed safety question.

It also states:

(b) ...The licensee shall also maintain records of tests and experiments carried out pursuant to paragraph (a) of this section. These records shall include a written safety evaluation which provides the bases for the determination that the change, test or experiment does not involve an unreviewed safety question. The licensee shall furnish to (the NRC)..., annually..., a report containing a brief description of such changes, tests, and experiments, including a summary of the safety evaluation of each.

Unless otherwise noted, each safety evaluation concluded that the subject change, test or experiment did not:

- Increase the probability of occurrence or the consequences of an accident or malfunction of structures, systems, or components important to safety previously identified in the FSAR;

- Create the possibility of an accident of or malfunction of a different type than any previously evaluated in the FSAR;

- Reduce the margin of safety as defined in the basis for Technical Specifications;

And therefore, do not involve an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-84-100

JAF-SE-84-139, Rev. 0 REPLACEMENT OF A HPCI TURBINE STEAM SUPPLY DRAIN POT LEVEL SWITCH

This modification consisted of the replacement of a HPCI Turbine Steam Supply Drain POT level switch, 23LS-90, including float chamber, switch assembly, and switch housing.

Summary - There were no changes to the FSAR or to the Technical Specifications as a result of this modification. The new level switch has the same performance characteristics as the original unit, and was purchased to the same specification as the original component. The new level switch was installed using an approved welding procedure. The I&C Department performed a calibration of the completed installation prior to returning the unit to service.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-82-017

JAF-SE-86-063 CORE SPRAY AND SAFE-END REPLACEMENT

The modification was a contingency dependent upon the findings of the JAF Inservice Inspection Program. If indications of major cracking were detected in the Core Spray stainless steel piping and/or safe-end welds, replacement activities or weld overlay repairs were to be performed. It was not the intent of this Safety Evaluation to address weld overlay repairs. This safety evaluation applied to the replacement of the Core Spray "A" and "B" loop stainless steel safe-ends, associated thermal sleeves and piping out to the first manual isolation valve for each loop (14CSP-14 A/B). The extent of the replacement was to be determined by the location(s) of weld defect(s). This modification basically replaced pipe prone to cracking with better material.

The incorporation of this modification:

There was no change in the operation or design basis as described in the Safety Analysis Report of the core spray system due to this modification.

This modification did not change the operation or design basis as described in the Safety Analysis Report of the core spray system.

All material design, fabrication, inspection, examination, and testing for this modification met or as in most cases exceeded the original construction code requirements. The replacement material had higher allowable stress values and better resistance to stress corrosion cracking.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-88-122

JAF-SE-86-215, Rev. 1 REACTOR VESSEL SHROUD HEAD BOLT REPLACEMENT

Cracking of shroud head bolt has been detected in most BWR's of similar design and age as FitzPatrick. Based on the study performed by G.E. on water chemistry trends, cracking is expected. G.E. recommended, in Service Information Letter (SIL) 433, performance of UT inspection on all the SHB's and the replacement of those found to be cracked.

Ultrasonic inspection (UT) was performed on all 36 shroud head bolt (SHB) during the 1987 refueling outage, a total of fifteen bolts were determined to be cracked and three bolts showed no back reflection and were characterized as suspect. As detailed in reference 7, an analysis of the most severe accident conditions shows that a minimum of twelve SHB's are required to assure structural integrity and to prevent movement of the shroud head/separator assembly. This modification involved the replacement of all cracked bolts.

Both G.E. and Kraftwerk Union (KWU) SHB's are designed to meet reactor vessel design pressure and temperatures (1250 psig and 575°F). A stress analysis was performed to verify that the resulting stresses of the new bolts would not be different from the original design during normal operating and transient conditions. The integrity of the reactor coolant pressure boundary was not affected by the replacement of the SHB's. This new design assured moisture separator to shroud mating since it conformed t the interface requirements between the shroud head and the moisture separator. This replacement did not alter the pressure boundary portions of the reactor vessel and did result in materials with improved properties in the non-pressure boundary areas.

The safety evaluation concluded that the modification did not involve a change in the Technical Specifications or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-87-031

JAF-SE-87-030 HELIUM LEAK DETECTION SYSTEM FOR IDENTIFYING MAIN CONDENSER IN-LEAKAGE

The reason for this modification was to provide a method for finding main condenser air in-leakage sources and waterside tube leaks thereby increasing therefore increase plant efficiency by reducing the leakage of air or water into the main condenser. Helium was added in a systematic manner to the external area of the main condenser and connected systems. The helium was pulled through any breaches in the system by condenser vacuum and eventually sensed by the detection unit. By coordinating the helium addition points and detection, breaches to the condenser system was identified and eliminated through maintenance actions.

Sections 11.4 and 16.5 of the FSAR were reviewed and it was determined this mod did not change the purpose and function described in that document. The sample pump and/or its function are not discussed in the FSAR or JAF Technical Specifications.

This modification was initiated to provide a permanent installation for the Helium Leak Detection equipment and associated components and eliminate the potential for leaks and spills associated with the present temporary installation of plastic tubing and temporary connections. The permanent installation did not pose a safety concern since it will not adversely affect plant equipment.

The addition of helium into the main condenser during testing had no detrimental effect on plant operations or performance due to the very small amount of gas introduced and the fact that helium is an inert gas and so will not chemically react with other materials present in the systems.

Helium leak detection has been used at nuclear and fossil power plants for over ten years. This type of helium leak test has been conducted at the Fitzpatrick Plant in the past using the proposed equipment in the same configuration and the same tie in point as the modification.

The installation of this modification did not adversely impact any associated systems or components. The additional weight of the tubing and components of this modification has been evaluated and did not affect existing or interfacing equipment. No safety-related equipment is in the area.

MODIFICATION: M1-86-123

JAF-SE-87-121, Rev. 2 REPLACEMENT OF HPCI ROBERTSHAW LEVEL SWITCHES (23LS-91A&B, 23LS-74A&B, AND 23LS-75A&B) WITH MAGNETROL LEVEL SWITCHES

This modification allowed the replacement of six existing Robertshaw control Company level switches with six Magnetrol International Inc. level switches. The plant components involved were 23LS-74A and 75A (HPCI - Condensate Storage Tank "A" level switches), 23LS-74B and 75B (HPCI - Condensate Storage Tank "B" level switches), and 23LS-91A&B (HPCI - suppression chamber level switches). An additional support was installed on the drain line between Torus penetration X-206C and wall sleeve S-206C in the Torus Room to resolve WR #23/063528.

This level switch replacement did not affect the safety design bases of the Emergency Core Cooling Systems (ECCS) controls and instrumentation and did not degrade the operability of the HPCI System. The replacement Magnetrol level switches were similar in design as the Robertshaw switches. Magnetrol International could provide a switch to satisfy 10CFR50, Appendix B, requirements. Installation of the new level switches provided the plant with a reliable switch with replaceable parts.

The safety evaluation concluded that the modification did not involve a change in the Technical Specifications or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-88-096

JAF-SE-88-072, Rev. 0 MAIN STEAM ISOL ION VALVE 19AOV-80A/B/C/D AND 29AOV-86A/B/C/D ACTUATOR REBUILD AND UPGRADE

The Main Steam Isolation Valve (MSIV) actuator manufacturer, R.A. Hiller Company, have modified several subcomponents of the actuators to replace items which are no longer manufactured and to improve the actuator design based on service experience. The JAF Maintenance Department, as part of actuator preventive maintenance, has rebuilt the actuators to replace these subcomponents and upgrade the actuators to the manufacturer's current design.

ITEM	DESCRIPTION OF CHANGE
Air cylinder head and end cap (SA-AO11, Items 1B & 2B)	Air ports and mounting hole configuration modified to accept new air control package
Hydraulic cylinder cap and cartridge bushing (SA-AO11, Items 2A & 21)	Rod packing chamber size modified for new packing system using less packing rings.
Piston Rod (SA-AO11, Item 3A)	Piston rod material changed from carbon steel to chrome plated stainless steel to improve corrosion resistance of exposed rod.
Air Control Assembly (SA-AO12)	Assembly redesigned to improve layout and increase ease of access to sub-components. All electrical components certified to meet EQ requirements. Quick disconnects incorporated to make replacement of solenoid valves easier. Stainless steel pneumatic fittings provided for superior corrosion resistance. Air control valves modified to replace obsolete parts (no longer manufactured) with current design.
Hydraulic Fluid	Changed from Shell Iris 902 (Mineral Base) to GE SF1147 (Silicon Base).
Fasteners (SA-AO11, Items 71, 73, 75, 83-85)	Fastener design modified for new hydraulic flow control valves and to incorporate lockwashers to prevent loosening of fasteners.
Elastomers (SA-AO11, Items 11, 16, 22, 76, 77, 86-88)	Elastomer material standardized as Viton.

Paint	Steel (Non Stainless Steel) parts painted with epoxy paint for improved corrosion resistance.
Lifting Eye	Lifting eye added to actuator to improve ease of handling.

This modification, in conjunction with modification M1-87-059, upgraded the actuators on all MSIV's (29AOV-80A-D and 29AOV-86A-D).

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-87-163

JAF-SE-89-050, Rev. 2 JET PUMP INSTRUMENT NOZZLE MODIFICATION

Prevention of Intergranular Stress Corrosion Cracking (IGSCC) is required of plants like JAF using 304 series stainless steels. An accepted approach for preventing IGSCC is to reduce the dissolved oxygen content in the reactor cooling system water, by implementing Hydrogen Water Chemistry (HWC). NYPA has implemented HWC at JAF. However, water in the Jet Pump Instrument (JPI) Nozzles was normally stagnant, and did not receive the full benefit of JAF's HWC system. Therefore, to provide HWC protection against IGSCC in the nozzle areas, small lines were installed to provide the needed flow through the nozzles.

The modification's only safety-related function is to maintain pressure integrity. This function is assured, because the modification is supplied to the same or more stringent requirements than the FSAR requirements for the Recirculation System and RCPB.

The modification (a) was supplied to the same or more stringent requirements than those required in the FSAR, (b) will prevent IGSCC, (c) did not change any inputs to a FSAR transient or accident analysis, and (d) and did not impact any radiation dose calculations in the FSAR.

The modification did not introduce any new failure mode or component/system interaction.

This modification did not impact the Security Plant or the Fire Protection System, and was provided in accordance with the Quality Assurance Program.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION:

JAF-SE-89-091 HEAVY LOAD ANALYSIS FOR OUTBOARD MSIV ACTUATORS (29AOV-86(OP)A-D)

This Safety Evaluation performed a heavy load lifting analysis for installation and removal of Main Steam Isolation Valve (MSIV) actuators from the outboard MSIVs.

This heavy lift did not create the possibility of an accident or malfunction of a type different than any evaluated previously in the FSAR because the controls imposed on the lift prevented damage to any plant equipment.

The lift met all controls specified in NUREG 0612 to prevent damage to equipment required for a safe shutdown.

The controls in place of the lift ensured no damage occurs to plant components which could affect these items.

This heavy load life did not affect the environmental impact of the plant or involve an unreviewed environmental question because these items were not applicable to heavy lifts of components.

The safety evaluation concluded that this action did not involve a change in the Technical Specifications or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-88-007

JAF-SE-90-001 DEMOLITION OF RADWASTE LAUNDRY DRAIN PUMPS

The purpose of this modification is to remove a portion of the existing Detergent Waste (Laundry) Drain System that had been retired in place since September of 1987. The portion removed consisted of the Laundry Drain Pumps (20P-26A and 20P-26B), the Laundry Drain Pumps (20P-26A and 20P-26B), the Laundry Drain Filter (20F-27) and associated piping. The space provided by the removal of the above equipment will be utilized for the installation of the oil/water separator. The removal of the above equipment also prohibits the inadvertent discharge of Laundry Drain Tank contents to the discharge canal. This modification was terminated upon the completion of modification F1-88-007B (Oil/Water Separator Installation).

Removal of both of the existing Laundry Drain Pumps (20P-26A and 20P-26B), the existing Laundry Drain Filter (20F-27) and the demolition of associated valves, piping, supports and instrumentation did not affect Radwaste System operation or safety since the Laundry Drain System has been retired in place as of September 1987 when all laundry began to be shipped off-site.

The safety evaluation concluded that the modification did not involve a change in the Technical Specifications or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-88-007

JAF-SE-90-002 INSTALLATION OF AN OIL/WATER SEPARATOR IN THE LIQUID RADIOACTIVE WASTE SYSTEM

The purpose of this modification was to reduce the concentration of oil/total organic carbon (TOC) in the low purity liquid waste associated with the Floor Drain Subsystem. The liquid waste in the Floor Drain Collector Tank (20TK-28) was allowed to drain to the Radwaste Building Floor Drain Sump (20TK-218) and was then transferred to the Waste Neutralizer Tanks (20TK-642A&B). These liquid wastes generally have low radioactivity concentrations and shall now be collected in the Floor Drain Collector Tank (20TK-28) and then discharged to the Radwaste Building Equipment Drain Sump (20TK-234) for subsequent processing. Oil/TOC in the liquid waste has a tendency to foul the resins used in radioactive waste processing, resulting in lower efficiency. Oil/TOC have also been known to interfere with filtration, heat transfer, and evaporation processes which then results in the carryover of TOC into the evaporator distillate.

This modification to the Liquid Radioactive Waste Systems is non-safety related, as there are no safety systems and/or equipment connected to or associated with the Radwaste System, or even located within the Radwaste Building. The modification does not appreciably alter the current operating philosophy associated with the liquid radwaste system, since the separator will simply be an intermediate step in the transfer of low purity liquid waste from the Floor Drain Collector Tank (20TK-28) to the Radwaste Building Equipment Drain Sump (20TK-234)

MODIFICATION: M1-87-026

JAF-SE-90-015 23MOV-19 AND 23MOV-20 ACTUATOR REPLACEMENT

This modification to the HPCI pump discharge valve 23MOV-20 and HPCI injection valve 23MOV-19 replaced the valve actuators.

The replacement operators simplified maintenance and spare parts requirements. The replacement operators were purchased to specifications which met or exceeded the original specification requirements. The operators were environmentally qualified to the requirements of 10CFR50.49.

The existing valve stem and stem nut were replaced to maintain the required stroke time. An analysis of the increased weight of the actuator on the valve yoke and body and associated piping had been performed and found to be acceptable and where necessary the existing supports have been modified to meet the new piping loads.

Based on the safe load travel path planned for these actuators over the Torus Room catwalk, it was determined that this modification complied with the requirements of NUREG 0612. To facilitate removal and reinstallation of the operators, a separate rigging beam for each valve location had been designed in accordance with AISC Manual of Steel Construction, 9th edition.

The normal function, safety function, failure mode, and operating times for the valve was not changed by this modification, and therefore, the operation of the HPCI System was not affected.

The safety evaluation concluded that the modification did not involve a change in the Technical Specifications or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-90-002

JAF-SE-90-019 OFFGAS ACTION PLAN

This modification was performed on the Off-Gas system to make improvements to system reliability and to reduce the MAN-REM exposure during operations. The following modifications were grouped together into one modification package to coordinate the work effort to be performed on the Off-Gas system.

Off-Gas Air Dryer Regeneration Time

This modification changed the Off-Gas dryer regeneration time to reduce moisture in the off-gas system, a source of activated charcoal bed trips.

Isolation of 01-107R-4B Recombiner and 01-107PH-8B Preheater

The idle "B" Recombiner train is one possible path for off gas leakage. Operating experience has shown the "B" recombiner train to be unnecessary for operations. This system was permanently isolated by cutting and capping the pipe to this train.

Instrument Air Purge Line

This modification relocated the instrument air purge isolation valves to the east electric bay. This modification allowed isolation of the instrument air to the off-gas system without entering the Condenser Bay.

O_2 Supply Isolation Valve

This modification installed an isolation valve in the oxygen system adjacent to the "B" condenser air removal pump. This modification allowed isolation of the oxygen supply to the off-gas system without entering the Condenser Bay.

Heat Trace Recombiner 01-107R-4A

The purpose of this modification was to install a heat trace system on the Off Gas Recombiner 01-107R-4A tank to decrease the normal warmup time required for initial operation of the Off Gas System. This modification will preheat the catalyst and tank assembly to a temperature of 190°F ±10°F. Warm up time is based on a 12 hour preheat time and control of the tank shell temperature within a predefined maximum surface temperature. The heat trace element surface temperature shall not exceed 700 ± 50°F during the warmup procedure.

MODIFICATION: M1-90-002
JAF-SE-90-019 OFFGAS ACTION PLAN
(continued)

The operation of the off gas system remained unchanged. This modification enhanced the reliability of the off gas system by minimizing the possibility of off-gas leakage through the "B" train, by reducing the moisture content of the off gas and by allowing the recombiner to be placed into service prior to 90% power. This modification also minimized operator exposure by the installation of an oxygen isolation valve and the relocation of the instrument air purge isolation valve outside of the recombiner area.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-90-006

JAF-SE-90-023, Rev. 1 REMOVAL OF RWCU OUTBOARD ISOLATION BYPASS VALVE 12MOV-80

This modification permanently removed and deleted the Reactor Water Cleanup System (RWCU) outboard containment isolation bypass valve (CIV) 12MOV-80 with associated electrical and control equipment and cables/wires. This valve failed to cycle properly and was declared inoperable. This valve had been disabled in the closed position for containment isolation.

The CIV bypass valve 12MOV-80 was a 1" bypass around the 6" main RWCU CIV 12MOV-18. This 1" bypass was intended to allow operational control of pressurization and heat up of the RWCU system.

The intended warm-up function of the 12MOV-80 valve has been replaced by alternate system lineups which more effectively perform the function of the 12MOV-80 valve.

The functions of the valve were to provide containment isolation, isolation on a standby liquid control system injection signal, and a reactor coolant pressure boundary. 12MOV-80 was removed and two LLRT connections were added at the weld-o-lets where the bypass line was previously connected. The two LLRT connections and the existing 6" RWCU line has been seismically analyzed and meet the applicable codes and standards for the application.

The removal of 12MOV-80 reduces the DC load on the emergency power supplies required for containment isolation, and reduces the risk of a failure of containment isolation at the location. The normally closed LLRT connections do not require action for containment isolation.

The existing HELB leak detection system for RWCU provides adequate coverage in the area of the modification to provide early indication of a potential leak. This modification did not adversely affect this leak detection system.

Containment isolation valves are listed in AP-1.16 and FSAR Table 7.3-1. AP-1.16 and FSAR Table 7.3-1 was revised to delete the 12MOV-80 valve.

The proper QA requirements have been met for the preparation of documents, installation and procurement of materials for the modification.

MODIFICATION: M1-90-085

JAF-SE-90-063, Rev. 1 REPLACEMENT OF SERVICE WATER TO RHR-SW KEEP-FULL CHECK VALVE 10RHR-431A AND 431B

This modification to the RHR-Service Water Keep-Full system replaced valves 1-RHR-431A and 431B. These check valves had worn seats and pistons due to aging and no longer operate properly. These valves are located in Service Water Keep-Full line to "A" and "B" RHR-Service Water Discharge Piping. New branch lines were added with valves to provide for future testing of each check valve.

This modification replaced the horizontal type check valves and relocated from a vertical location to a horizontal location. Additional test connections with isolation valves were installed.

The new valves provide better shut-off capabilities than the original valves and were purchased to specifications which met or exceeded the original specification requirements.

The normal function of the valves were not changed by this modification and therefore the operation of the Service water to RHR Service Water was not impacted.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-88-039

JAF-SE-90-070, Rev. 0 REPLACEMENT OF CO_2 DAMPERS IN THE DIESEL GENERATOR SWITCHGEAR ROOMS

This modification replaced the existing six horizontally mounted, curtain type dampers, located in the EDG switchgear rooms, with louver type dampers. Two dampers are in the Turbine Building Ventilation System (Sys. No. 92). These curtain type dampers had become unreliable and resulted in continuous maintenance problems. The present arrangement closes the damper via a piston type actuator located inside the duct. These dampers are tested every six (6) months. The present position indicators and actuators cause jamming of the damper which in turn prevents the complete closure of the damper.

This modification installed a louver, spring closed, UL Listed and three-hour fire rated damper. This design met all the above-mentioned purposes and enhances the efficiency of the CO_2 system by achieving a lower leakage rate. The design and installation complies with relevant codes, standards and criteria.

The original design basis of the Fire Protection System as stated in JAF FSAR Section 9.8 was not changed and the replacement dampers met or exceeded all applicable design requirements of the original dampers.

The design basis for the Fire Protection System as stated in FSAR Section 9.8 is:

> The design basis for the Fire Protection System is to prevent, detect and suppress any single fire or probable combination of simultaneous fires which might occur.

The modification was evaluated for postulated equipment failures and no new type of scenario exists.

The design function of the dampers was not changed and the replacement dampers met or exceeded all requirements in performing that design function.

MODIFICATION: F1-88-178

JAF-SE-90-086, Rev. 1 REPLACEMENT OF VALVES 20MOV-82, 20AOV-83 AND 20AOV-95

The purpose of this modification was to improve the leak tightness of the Primary Containment by replacing valves whose past LLRT test results have indicated poor performance.

This modification replaced the existing inboard containment isolation valve 20MOV-82 and outboard isolation valves 20AOV-83 and 20AOV-95 with new replacement valves having improved design features and with better shut-off characteristics.

The new valves were purchased to specifications which meet or exceed the original specification requirements.

The normal function, safety function and failure mode of the valves were not changed by this modification and therefore the operation of the Radwaste and Primary Containments Systems were not affected.

Based on the review performed herein by Nuclear Engineering and Design, the following was concluded for the Safety Evaluation:

The original design basis was not changed.

The new valves were purchased to specifications which met or exceeded the original specification requirements. The piping and supports were designed in accordance with the appropriate codes and standards and the piping maintained its pressure boundary under all conditions of operation.

This modification did not affect the Technical Specifications or the safety analysis of the plant as described in the JAFNPP FSAR Chapter 14.

The proposed modification did not degrade the Security Plan, Quality Assurance Program or the Fire Protection System. This modification did not affect the Fire Protection System, it did not impact any safe shutdown components or fire barriers and it did not increase the combustible levels in any of the plant areas. This modification was implemented in accordance with the site's Quality Assurance program.

The proposed modification did not affect the environmental impact of the plant or involve an unreviewed environmental question, because this modification did not discharge any effluents.

MODIFICATION: F1-90-195

JAF-SE-90-087, Rev. 1 **REPLACEMENT OF REACTOR RECIRCULATION WATER SAMPLE LINE CONTAINMENT ISOLATION VALVES 02-2SOV-39 AND 40 WITH 02-2AOV-39 AND 40**

The purpose of this modification was to replace Reactor Recirculation Water Sample Line Containment Isolation Valves 02-2SOV-39 (Line No. 1"-WH-1504-14) and 02-2SOV-40 (Line No. 3/4"-WH-1504-14), at Penetration X-41, with new air operated valves in the existing location of the SOVs. This modification also restored line 3/4"-WH-1504-14, which was cut and capped in the Reactor Building, (Fuel Pool Heat Exchanger Room, El. 326') to its original design configuration so that the flow to the Crack Arrest Verification (CAV) System and the Sample Sink was restored.

All work was designated as QA Cat. I, with the exception of piping downstream of 02-2AOV-40 and air supply line to 02-2AOV-40, which were designated as QA Cat. II/III.

This modification replaced existing containment isolation valves with new air operated valves that met the LLRT criteria of 10CFR50, Appendix J.

The modification provides a valve design which is more appropriate for the application.

All the system components remain the same functionally.

The existing criteria have been used for the purchase and installation of the new valves.

This modification adheres to all applicable codes and standards for the purchasing and installation of the valves.

This modification replaces two existing valves and restores the sample line which does not alter any accident analysis currently addressed in the FSAR.

MODIFICATION: F1-90-196

JAF-SE-90-090, Rev. 2 REPLACEMENT OF REACTOR VESSEL BOTTOM DRAIN ISOLATION VALVE 02-3NBI-81

The purpose of this modification to the Nuclear Boiler Vessel Instruments System was to replace the existing Reactor Vessel Bottom Drain Isolation valve. The valve was replaced because of maintenance problems associated with the existing Velan globe valve.

This modification replaced the existing valve 02-3NBI-81, a globe valve, with a double disc gate valve.

The new valve has better design features than the original valve and was purchased to specifications which meet or exceed the original specification requirements. The function of the valve was not changed by this modification and therefore the operation of the Nuclear Boiler Vessel Instruments System were not affected.

Based on the review performed by Nuclear Engineering and Design, the following was concluded for the Safety Evaluation:

The original design basis was not changed.

The new valve was purchased to specifications which meet or exceed the original specification requirements. The piping and supports were designed in accordance with the appropriate codes and standards and the piping will maintain its pressure boundary under all conditions of operation.

This modification did not affect the Technical Specifications or the safety analysis of the plant as described in the JAFNPP FSAR Chapter 14.

This modification did not discharge any effluents.

MODIFICATION: F1-90-197

JAF-SE-90-091, Rev. 0 REPLACEMENT OF MOTOR OPERATOR FOR RCIC TURBINE STEAM SUPPLY OUTBOARD ISOLATION VALVE 13MOV-16

The purpose of this modification was to improve the stroke time of RCIC Turbine Steam Supply Outboard Isolation valve so that it would adequately meet the Technical Specification stroke time requirements by replacing the existing operator with a new larger operator of a similar type.

This modification to the RCIC Steam Supply Outboard Isolation Valve 13MOV-16 replaced the operator and modified the valve to receive the new operator. The stroke time will decrease in order to meet Technical Specification requirements and the motor torque available to operate the valve increased. The normal function, safety function and the failure mode did not change.

The following conclusions have been made:

The valve being modified (13MOV-16) is not an initiator of an accident evaluated previously in the FSAR.

The ability of valve 13MOV-16 to close when an isolation signal is received is not degraded.

The only equipment impacted is valve 13MOV-16 which is being upgraded to provide increased safety margins.

The failure mode of valve 13MOV-16 was not changed. If the valve 13MOV-16 fails to close due to a single active failure, redundant valve 13MOV-15 will close to provide containment isolation at Penetration X-10.

Changing the operator on valve 13MOV-16 did not create new types of failure modes.

By providing a larger operator and changing the gear ratio to adequately meet the required stroke time of 12 seconds, the margin of safety associated with the basis for closure time was not reduced.

This modification did not discharge any effluents.

MODIFICATION: M1-88-140

JAF-SE-90-092, Rev. 0 ON LINE REMOVAL OF EXCITER SLIP-RING BRUSHES FOR MAIN EXCITER

To install removable cartridge brushholders which allow the brushes to be changed on the main generator exciter while it is operating. At the present time the turbine must be shut down to change the exciter brushes.

This modification did not affect the safety analysis of the FSAR. The installation was not Category I and was performed based on the following conclusions.

The result of failure of the new design is the same as that of the present design; a turbine trip, and this event was already considered and analyzed.

The only consequence of any failure, including a failure while the brushes were being changed was a turbine trip and this event had already been considered and analyzed.

The modification is not safety related and the exciter was not addressed by the Technical Specification.

The exciter brushes do not affect the environment.

This modification did not change any safety-related systems and codes to which these systems are built to, did not involve any unreviewed safety questions and did not require any changes in the Technical Specifications.

MODIFICATION: F1-90-193

JAF-SE-90-101, Rev. 1 **REPLACEMENT OF INSIDE MAIN STEAM DRAIN OUTBOARD ISOLATION VALVE, 29MOV-77, OUTSIDE MAIN STEAM LINE DRAIN VALVE, 29MOV-78, AND MAIN STEAM LINE DRAIN TO CONDENSER VALVE, 29MOV-79**

The purpose of this modification was to reduce leakage from the Primary Containment by replacing containment isolation valve, 29MOV-77. Valves 29MOV-78 and 29MOV-79, which have required excessive maintenance were also replaced by this modification.

This modification replaced the existing outboard containment isolation valve 29MOV-77 and valves 29MOV-78 and 29MOV-79 with new replacement valves of a similar type with better shut-off characteristics and design features.

Based on the review performed by Nuclear Engineering and Design, the following was concluded for the Safety Evaluation:

The original design basis were not changed.

The new valve was purchased to specifications which met or exceed the original specification requirements. The piping and supports were designed in accordance with the appropriate codes and standards and the piping will maintain its pressure boundary under all conditions of operation.

This modification did not affect the Technical Specifications or the safety analysis of the plant as described in the JAFNPP FSAR Chapter 14.

This modification did not discharge any effluents.

MODIFICATION: M1-88-214

JAF-SE-90-103, Rev. 0 INSTALLATION OF STANDBY LIQUID CONTROL (SLC) SYSTEM DRAIN TANK AND ASSOCIATED COMPONENTS

The purpose of this modification was to document the technical and safety implications of the permanent installation of the 350 gallon Standby Liquid Control (SLC) drain tank and agitator, tank discharge pump, and associated mechanical and electrical components located on elevation 300' in the Reactor Building N.W. installed prior to January 1988. This modification involved new mechanical and electrical work to permanently install existing temporary equipment.

The drain equipment is QA Category II/III and did not adversely affect the safety objective and safety design bases as described in JAF FSAR Section 3.9. The drain equipment and their operation have no safety-related functions associated with them or JAF Technical Specifications are not affected by this modification. In conclusion, the SLC drain tank and associated equipment did not conflict with the safety objective and safety design basis of the Standby Liquid Control System as stated in the FSAR, nor result in changes to the Technical Specifications, nor effect any safety-related structures, systems, or components.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-88-069

JAF-SE-90-108 REPLACEMENT OF SUPPRESSION POOL COOLING ISOLATION VALVE 10MOV-39B

The purpose of this modification was to reduce leakage of the Primary Containment by replacing valves whose past LLRT test results have been poor.

This modification replaced the existing suppression pool cooling isolation valve 10MOV-39B with a new replacement valve of a similar type with better shut-off characteristics.

This plant modification to the Residual Heat Removal System was classified as Nuclear Safety Related, QA Category I and Seismic Class I.

The replacement valve was purchased to specifications which met or exceeded the original specification requirements. The valve is designed to ANSI B31.1 and meets all of the design conditions for ANSI B16.34 for special class valves. The subject valve operator is environmentally and seismically qualified to IEEE Standards 323, 344 and 382 for use inside containment.

The normal function, safety function, failure mode, and operating times of the valve were not change by this modification and therefore the operation of the Residual Heat Removal and Primary Containment systems were not affected

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-89-044

JAF-SE-90-110 REPLACE TURBINE GENERATOR COUPLING BOLTS

The purpose of this modification was to remove the studs and nuts from the turbine generator coupling bolts A B and C couplings and replace them with a new stud design that takes less time to assemble and disassemble. The new design is available from G.E. as a retrofit.

This modification did not affect the safety analysis of the FSAR. The equipment modified was not Category I and this modification was installed based on the following conclusions.

The result of any failure of the design will be the same as that of the present one: The turbine will trip. This event has already been considered and analyzed.

The only result of a stud failure for the present design and for the proposed design is a turbine trip and this event has already been considered and analyzed.

The modification was not safety related and did not affect systems addressed in the Technical specifications.

MODIFICATION: M1-89-065

JAF-SE-90-112 67TCV-119 AND 120 VALVE AND ASSOCIATED STRAINER REMOVAL

The purpose of this plant modification was to remove temporary control valves 67TCV-119 and 67TCV-120 and associated strainers and to replace them with a spool piece of stainless steel pipe. This action was required by JAF commitments to the NRC contained in letter JPN-91-047 to address the concerns of air-operated valves which fail closed rather than open.

Section 12.2.2 of the FSAR states that the Emergency Service Water System is a QA Category I system. Therefore, this piping was also seismically supported. This modification maintained these piping qualifications.

This modification improved the operability of the plant equipment in the electric bay by enhancing the operability of unit coolers 67UC-16A and B.

The safety evaluation concluded that the modification did not involve a change in the Technical Specifications or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-88-007 TEST: POT-20AA

JAF-SE-91-001 PREOPERATIONAL TEST NO. 20AA OIL/WATER SEPARATOR MODIFICATION

This preoperational test was conducted to demonstrate the proper functioning of valve, instrumentation, and equipment components associated with Plant Modification F1-88-007 (Oil/Water Separator Modification). Modification F1-88-007 includes the installation of a coalescing-type oil/water separator and associated piping on elevation 250'-0" of the Radwaste Building. The purpose of the oil/water separator was to reduce the Oil/Total Organic Carbon (TOC) in the liquid effluent from the floor drain collector tank (20TK-28) before it entered the radwaste building equipment drain sump (20TKS-234) for subsequent processing by the radwaste evaporators. Associated with the separator addition, a tie line between the equipment drain sump pump (20P-8) discharge line and the main 4 inch header to the waste neutralizer tanks (20TK-642A & B) was added. The existing waste sludge pump, suction and discharge lines were replaced with a new heavy-duty sludge pump and larger diameter lines. The new equipment was located within the Radwaste Building on elevation 250 feet and the controls are mounted on Panel 25-17 in the Radwaste Control Room.

The conductance of the preoperational test which tested the oil/water separator in the floor drain subsystem of the Radioactive Liquid Waste System did not impact on any safety related or environmentally qualified components or systems, or affect overall plant safety.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specifications or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-90-203

JAF-SE-91-002, Rev. 1 RWCU SYSTEM IMPROVEMENTS - MANUAL ISOLATION VALVE REPLACEMENT (VALVES 12RWC-19B, 29B AND 73B

The purpose of this modification was to replace the "B" side Reactor Water Cleanup Pump suction and discharge, manually operated valves to provide better isolation during pump maintenance. The existing valves have been difficult to operate, requiring excessive force, which has resulted in bent valve stems.

This plant modification to the Reactor Water Cleanup System is classified as QA Category II/III and Seismic Class I.

The replacement valves have been purchased to specifications which met or exceeded the original specification requirements. The valves were designed in accordance with the original design code ANSI B31.1 and met all of the design conditions for ANSI B16.34 for special class valves.

The valve function, and design will not be changed by this modification and, therefore, the operation of the Reactor Water Cleanup System will not be impacted.

The safety evaluation concluded that the modification did not involve a change in the Technical Specifications or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-87-049

JAF-SE-91-007, Rev 1 REPLACEMENT OF REACTOR VESSEL HEAD VENT VALVES 02AOV-17 AND 02AOV-18

The purpose of this modification was to replace reactor vessel head vent valves whose past performance has been poor and has required excessive maintenance.

This modification replaced the existing reactor vessel head valves 02AOV-17 and 18 with new valves having improved design features and with better shut-off characteristics.

This plant modification to the Nuclear Boiler System was classified as Nuclear Safety Related, QA Category I and Seismic Class I. The modification to the Drywell Cooling and Ventilation System is QA Category II/III.

The replacement valve was purchased to specifications which meet or exceed the original specification requirements. Valves 02AOV-17 and 18 are designed in accordance with the 1989 Edition of ASME Section III, Class 1.

The operation of the Nuclear Boiler System was not affected by this modification because the design, function, failure mode, and operating time of the valves was not changed by this modification.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-81-018

JAF-SE-91-008 REPLACEMENT OF REBOILER STEAM SUPPLY "B" NON-RETURN VALVE 31NRV-117B (EXTRACTION STEAM SUPPLY TO THE REBOILER

The purpose of this modification was to replace reboiler steam supply "B" non-return valve 31NRV-117B whose past performance has been poor and has required excessive maintenance.

The air cylinder operator of the existing valve was fabricated from carbon steel materials. Condensation from the service air supply to the operator resulted in the formation of rust inside the operator and poor operator performance.

This modification replaced the existing 31NRV-117B with a new replacement valve of a similar type. Carbon steel parts of the new operator exposed to service air were fabricated from stainless steel materials.

This plant modification to the Extraction Steam System was classified QA Category II/III.

The replacement valve was purchased to specifications which meet or exceed the original specification requirements. The valve is designed in accordance with ANSI B31.1 and meets all of the design conditions for ANSI B16.34.

Valve function, failure mode and operation were not changed by this modification and therefore the operation of the Extraction Steam and Reboiler Systems was not affected.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-89-027

JAF-SE-91-009 REPLACEMENT OF RHR SHUTDOWN COOLING OUTBOARD ISOLATION VALVE 10MOV-17

The purpose of this modification was to reduce leakage of the Primary Containment by replacing valves whose past LLRT test results have been poor.

This modification replaced the existing RHR Shutdown Cooling Outboard Isolation valve 10MOV-17 with a new valve of a similar type with better isolation characteristics.

This plant modification to the Residual Heat Removal System was classified as Nuclear Safety Related, QA Category I and Seismic Class I.

The replacement valve was purchased to specifications which met or exceeded the original specification requirements. The valve is designed to ANSI B31.1 and met all of the design conditions for ANSI B16.34 for special class valves. The subject valve operator is environmentally and seismically qualified to IEEE Standards 323, 344 and 382.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-84-104

JAF-SE-91-010 JAF NPP FW HEATER(S) LEVEL AND PRESSURE INSTRUMENTATION

The purpose of this modification was to provide direct reading level and pressure instrumentation to determine condensate level and shell side pressure on feedwater heaters 33E-2A, 33E-2B, 33E-3A, 33E-3B, 33E-4A, 33E-4B, 33E-5A, 33E-5B, 33E-6A and 33E-6B. This was accomplished by installing direct reading level and pressure gauges on these feedwater heaters. These readings will be used to optimize feedwater heater output and overall plant performance.

The feedwater heaters are not safety-related. All work associated with the design, fabrication and installation was Q.A. Class II/III. Socket weld connections will be used on all piping with the exception of a threaded connection on the pressure gauge and each level gauge isolation valve.

The design pressures for the pressure gauges were selected to exceed the maximum design pressure at the individual heat exchanges. The accuracy is guaranteed to meet ANSI B40.1 grade 3A.

The operation of the feedwater heaters was not affected by this modification because the function and design of the feedwater heaters were not changed by this modification.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-90-187

JAF-SE-91-015, Rev. 1 EHC PUMPS DISCHARGE ISOLATION VALVE ADDITION, 94EHC-FV-48 AND 94EHC-FV-49

The purpose of this modification was to provide a block valve in each Electro Hydraulic Control EHC pump discharge line at the EHC skid and to facilitate pump maintenance during system operation. Only a check valve, 94EHC-FV-7(8), was provided in each line to supply isolation. The EHC skid was located at 252' elevation of the turbine building.

This modification also used to change the piping between the main EHC pumps (94P-7A and B) and relief valves 94FV-6 and 5 respectively.

This modification did not adversely affect Turbine Electro-Hydraulic System operation since the valve was manufactured to ANSI B16.34 requirements and installed (welded) to ANSI B 31.1 requirements and is used during EHC pump maintenance only. This system is a non-safety related system and changes to this system did not affect evaluations and results described in Chapter 14 of the FSAR.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-91-074

JAF-SE-91-024, Rev. 1 RECIRCULATION FLANGE ECP MONITORING ASSEMBLY

An Electrochemical Potential (ECP) Monitoring Assembly was added to the recirculation loop B suction piping, downstream from valve 02MOV-43B.

The purpose of the ECP Monitoring Assembly was to measure the electrochemical potential of the reactor water in a Hydrogen Water Chemistry (HWC) environment. This verified the Crack Arrest Verification System (CAVS) ECP sample representativeness and enabled fine tuning of the HWC system. It provided redundant ECP monitoring capability while operating with HWC in service, which serves as a benefit in light of recent failures of the existing CAVs sample isolation valves. Also, the modification provides ECP monitoring at all power levels versus only for power levels greater than 90% which was the present monitoring capability of CAVS.

Based on the Review and Analysis above, it was concluded that the addition of ECP monitoring assembly:

The ECP monitoring assembly was designed, fabricated, tested and installed in accordance with the ANSI B31.1-1967 requirements. In addition, the consequences of postulated failures are bounded by the current FSAR analyses, Sections 6.5.3.1 and 14.6.1.3.

The postulated failure modes i.e., pipe break, jet impingement, and drywell missiles were already considered in the design of JAFNPP. The consequences of postulated ECP monitoring assembly failures are bounded by the current FSAR analyses, Sections 6.5.3.1 and 14.6.1.3.

The leakage limits specified in Section 3.6.E of the Technical Specifications remain unchanged, therefore, there was no reduction in the margin of safety as defined in the Technical Specifications bases.

There was no change to the current standards and limits on effluents.

MODIFICATION: F1-89-141

JAF-SE-91-025, Rev. 0 DATA LOGGER/RECORDER REPLACEMENT

A number of JAFNPP recorders have become obsolete, making it difficult to obtain replacement parts. The purpose of this modification (reference V.1) was to restore and upgrade the recording and data logging functions of various Control Room and Auxiliary Boiler Room recorders.

The subject recorders were replaced with state of the art Westronics recorders, which allows ease of maintenance and more efficient operation.

This modification also:

- removed non-functioning trend recorders 71TR-1 and 71TR-2 from the Control Room Panels
- replaced Auxiliary Boiler flow and temperature sensing devices to allow a compatible interface with the new 87FR 100A & B recorders
- relocated a label plate from inside the cover of control room recorder 02-3LR-98 to the bezel of the recorder
- replaced two temperature selector switches 12TSS-128 and 12TSS-144 on Panel 09-21 which were exhibiting binding.

It can be concluded that the replacement of these recorders did not degrade the design bases or functions of plant equipment and were performed based on the following conclusions:

The new recorders, multiplexers, transducers and switches constitute an in-kind replacement of the equipment existing and met the design and qualification requirements of the existing equipment. This modification was an enhancement to existing equipment by removal and replacement of obsolete instrumentation per the following:

The installation of new recorders in the control room panels enhanced availability of the recorders and did not degrade the integrity of these panels.

All fire protection guidelines were followed, new fire hazards were not created.

Separation criteria for electrical cables inside the control panels were not disturbed since all new recorders in the JAFNPP Control Room were re-connected to the original instrument loops.

MODIFICATION: F1-89-141

JAF-SE-91-025, Rev. 0 DATA LOGGER/RECORDER REPLACEMENT
(continued)

This modification provides for an in-kind replacement of existing equipment without significant technical or functional changes.

This modification did not create any new releases or release pathways. The new equipment did not affect the existing plant design guidelines.

This modification did not interface with the security plant system and this modification did not involve pulling cables through, or breaching any fire barriers.

It is concluded that the removal of existing Control Room and Auxiliary Boiler Room recorders, associated multiplexers, transducers and switches and replacement of the same with new equipment as described above did not present an unreviewed safety question. This is based on review of the design specifications, FSAR and Technical Specifications. Also, the new recorders met all the criteria set forth by NUREG-0700 and JAFNPP procedures and specifications.

MODIFICATION: M1-91-004

JAF-SE-91-030, Rev. 1 CHEMICAL DECONTAMINATION - ADDITION OF PERMANENT PIPING AND VALVE - ADDITION OF CONCRETE BUILDING

Due to increasing shutdown radiation levels and resultant high man-rem exposures on outage jobs conducted in the drywell, there was a need to lower radiation levels at the JAF plant. One of the primary contributors to the exposure in BWR's was in the oxide layer of the stainless steel recirculation piping. Chemical decontamination is a known method to remove the oxide layer with minimal corrosion to the piping. This modification, which provided the required design changes to the plant to support chemical decontamination significantly reduced overall man-rem exposure for the 1992 Refueling outage.

Systems that were decontaminated are as follows:

a) Reactor water Recirculation system

b) Section of the Reactor Annulus to elevation 301'-5"

c) Jet Pump Instrument drain lines

d) Most of the RWCU Suction to the Pump and up to first isolation valve on discharge piping

e) RHR supply to the outboard isolation valve

f) RHR/Fuel pool cooling - cross-tie

The chemical decontamination process was not addressed in this modification, only the permanent modifications to the plant to support this process were addressed. The specific piping modifications allows for permanent connections on a) and d) described above to supply and return the decontamination solvents. This eliminated the need for installing temporary connections in subsequent outages during critical path time.

The Reactor Recirculation system including the affected branch lines is defined as QA Category I in accordance with JAF FSAR Section 12.2.3. The addition of piping and valves to RWR branch connections did alter or impact system operation. The newly installed valves remained closed and flanges remained blanked during normal plant operation.

Leak Testing of the new assemblies were performed in accordance with the JAF FSAR and applicable code requirements of ANSI B31.1 (1967) and ASME Section XI (1980 Edition thru winter 1981 addenda).

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-89-051

JAF-SE-91-031, Rev. 0 STACK DILUTION FAN MOTOR MATERIAL SUBSTITUTION

The purpose of this modification was to provide a dedicated spare motor for stack dilution fans in accordance with a change of the design requirements.

The existing (originally installed) motor was obsolete and no longer available from the equipment manufacturer (Joy Technologies) or from the motor manufacturer (Reliance Co.)

Sections 11.4 Gaseous Radioactive Waste System and 1.6.1.6 Radioactive Waste System of FSAR were reviewed. Technical Specifications were reviewed.

The replacement motor exceeded new area classification requirements of Class I Division 2 locations in accordance with NEC. The modification did not change the function or design of the stack dilution fans.

This modification did not affect the environmental impact of the plant, did not affect the EQ, Fire Protection, QA or Security Programs. There was no change in the FSAR and no reduction in the margin of safety.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-91-083

JAF-SE-91-033, Rev. 1 INSTALLATION OF A LIQUID RADWASTE FILTER/DEMINERALIZER SYSTEM IN THE "A" HOPPER ROOM OF THE RADWASTE BUILDING

This safety evaluation is intended to demonstrate that the modification to the Radioactive Liquid Waste System described by this evaluation did not affect overall plant safety or the environment. It also demonstrated that the modification complied with the overall system design criteria contained in the FSAR.

The purpose of this modification was to install mechanical and electrical services to support both the relocation of the temporary filter/demineralizer system from the East Truck Bay location to the "A" Hopper Room and the addition of a sluiced demineralizer system in the "A" Hopper Room. Both locations (East Truck Bay and "A" Hopper Room) are on elevation 272'0" of the Radwaste Building. The temporary filter/demineralizer system for liquid radwaste processing was added by Temporary Modification 91-040.

This modification, which provided, mechanical and electrical services to support the relocation of the temporary filter/demineralizer system and the addition of a temporary sluiced demineralization system, did not affect any safety related or environmentally qualified components or systems, or affect overall plant safety. In addition, the modification complied with the overall safety design bases contained in the FSAR.

Temporary filtration/demineralization and sluiced demineralization type liquid radwaste processing systems was located in the shielded "A" Hopper Room on elevation 272'-0" of the Radwaste Building. The room has controlled access and had appropriate radiological protection and monitoring. Design and installation complied with relevant codes, standards and criteria.

Based on the review performed herein, the following is concluded for this modification as identified in Section "A" of this Safety Evaluation:

This modification supported treatment of liquid waste in a manner not described in the FSAR. However, the Safety Design Bases in Sections 11.2.3 and 11.3.3 have been reviewed and this modification did not result in operation outside these bases.

The proposed modification was evaluated for postulated equipment failures and no new type of scenario exists. The modification was designed and installed in accordance with appropriate codes, standards, and plant procedures that provided a level of safety and personnel protection that is consistent with the radwaste system design basis.

JAF-SE-91-033, Rev. 1 INSTALLATION OF A LIQUID RADWASTE FILTER/DEMINERALIZER SYSTEM IN THE "A" HOPPER ROOM OF THE RADWASTE BUILDING
(continued)

This modification did not affect the Technical Specifications or the Safety Analysis of the plant as described in the JAFNPP FSAR Chapters 14 and 15. The Radioactive Liquid Waste System is a radioactive system, which processes radioactive liquids for subsequent concentration for burial.

This modification, did not alter the system or its operation in any way which increased the possibility of an unmonitored release or a release exceeding the limits specified in the Technical Specifications.

No additional fire load was added to the radwaste building by the installation of the processing system. The liquid waste treatment system is QA category II/III. Post installation checks (e.g., leak and operability tests) ensured the level of quality of the Radwaste System was not degraded.

The potential discharge of liquid radwaste system effluent to the environment was not affected by this modification.

The room has controlled access and had appropriate radiological protection and monitoring. Design and installation complied with relevant codes, standards and criteria.

MODIFICATION: M1-91-087

JAF-SE-91-036, Rev. 0   LPCI CHARGER/INVERTER CURRENT LIMITER

Correct an original design error in the LPCI charger current limiter circuit which caused the charger AC input current to increase above 100 amperes during recharge of the battery, rather than limit the current to 70 amperes maximum as called for in the UPS performance specifications. This modification to the current limiter problem was recommended by the vendor.

This modification ensured that the charger current limiter function on the LPCI UPS performs as required by the original design specifications. The only accident for which the LPCI UPS was required was the LOCA, FSAR chapter 14.6.1.3.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-91-048

JAF-SE-91-042, Rev. 0 EHC RECIRCULATING TANK VENT LINE TO THE EHC RESERVOIR (TIL #945-2)

The purposes of this modification was to add a vent from the EHC recirculating tank to the main EHC reservoir. This vent precluded the overpressurization and failure of the recirculating tank due to EHC transfer pump (94P-8) dead head conditions caused by relief valve failure and improper system valve lineups. The referenced Technical Information Letter 945-2 from General Electric states that this pump dead heads at 1000 PSIG. Another power plant has experienced such an EHC recirculating tank failure.

Turbine Generator Electro-Hydraulic Control System Operation was not degraded since the vent line was installed to the existing tubing specifications with the recommended swagelock fittings for this tubing. This system is a non-safety related system and changes to this system did not affect evaluation and results described in Chapter 14 of the FSAR.

This modification enhances the protection of the EHC Recirculating Tank 94TK-7 based on G.E. recommendations as stated in TIL 945-2.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-91-069

JAF-SE-91-043, Rev. 0 REPLACEMENT OF GE CR2820B TIME DELAY RELAYS WITH AGASTAT ETR TIME DELAY SERIES

To replace existing GE CR2820B time delay relays 10A-K93A (09-32 Panel) and 10A-K93B (09-33 Panel) with Agastat ETR Series relays per General Electric (GE) SIL 230, revision 2.

Installation and operation of the replacement Agastat ETR series relays for the GE CR2820B relays did not impact or affect operability of the RHR (LPCI Injection Mode) system or affect overall plant safety. The ETR series relays are designed as a direct functional and qualified replacement for the GE CR2820B series. The ETR series relays provide a higher degree of accuracy and reliability than the GE CR2820B series.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-90-238

JAF-SE-91-055, Rev. 1 — UPGRADE OF MONICORE PLUS NUCLEAR STEAM SUPPLY SYSTEM SOFTWARE TO 3D MONICORE NUCLEAR STEAM SUPPLY SYSTEM SOFTWARE, WITH THE REQUIRED UPGRADE OF THE MICROVAX II COMPUTER HARDWARE TO MICROVAX 3800 COMPUTER HARDWARE

The purpose of this modification was to replace the current NSSS core monitoring software (MONICORE Plus) with newer, more capable 3D MONICORE NSSS core monitoring software. 3D MONICORE provides three dimensional modelling of reactor thermal and nucleonic properties. It provided Reactor Engineers with greater predictive and analytic tools than the current software.

This modification to the EPIC system replaced the MONICORE Plus software and associated MicroVAX II computer hardware with 3D MONICORE software and MicroVAX 3800 computer hardware. The new software provides timely and comprehensive Nuclear Steam Supply System (NSSS) reports than the old NSSS software and will have continued support of the GE available. 3D MONICORE provides three dimensional model of the reactor thermal and nucleonic properties, reduces the uncertainty in the core monitoring functions, and provides the Reactor Engineer with a larger and more versatile suite of diagnostic and predictive tools for core performance evaluation. The operation of the EPIC system and the EPIC NSSS displays were not changed by this modification.

Based on the safety reviews performed for the EPIC system and a specific review of the software and hardware changes made within the EPIC system, the following was concluded for the Safety Evaluation.

The original design basis was not changed. The modification did not affect the functional operation, safety systems, or the instrumentation of the plant.

The modification did not affect the functional operation, safety system, or instrumentation of the plant. The new software and hardware installed, met or exceeded the original requirements of the system.

This modification did not affect the TS or the safety analysis of the plant as described in the FSAR Chapter 14. The modification increases the margins of safety by reducing the core monitoring uncertainty.

This modification did not impact any safe shutdown components, and did not increase the combustible loading in any of the plant areas.

MODIFICATION: F1-91-056

JAF-SE-91-056, Rev. 0   TURBINE BLDG - SHIELD WALL MODIFICATION

The purpose and objective of this modification was to substantially reduce the radiation dose rates in the existing facilities located east of the Turbine Building. This modification also reduced the radiation dose rate for the new Support and Administration Facility (Reference MOD No. F1-90-013). The targeted dose rate is 50 Micro R/Hour for areas of normal occupancy, based on ALARA analysis.

1. The Turbine Building and Heater Bay Roof were analyzed and qualified by Stone & Webster Engineering Corporation for the additional loads imposed by the new shield wall.

2. The new shield wall design was performed by NYPA. Design and analysis was preformed within the requirements of the applicable codes and standards. The new wall is supported by a Class II structure which is not safety related (JAF FSAR Section 12.2.4).

3. This modification provides an improved radiation protection for areas east of the Turbine Building. It also reduces the radiation dose rate for the new Support and Administration Facility, which has resulted in substantial reduction of the construction cost of this new building.

This modification did not change or alter the design of existing plant systems, portion of systems, or individual component.

This modification did not involve any change to existing safety or non-safety equipment.

None of the technical requirements were affected by this modification.

The materials used in the construction of the new wall did not alter the present environmental condition.

The modification did not change any of the detection or suppression systems or other installed equipment/components associated with the Fire Protection System and it did not affect any of the shutdown equipment/components used in the Appendix R analysis or modify any commitments or assumptions used in this analysis.

MODIFICATION: F1-88-127

JAF-SE-91-057, Rev. 0 DRYWELL CONTINUOUS AIRBORNE RADIOACTIVITY MONITOR (CAM) MODIFICATION (PHASE I)

This safety evaluation was intended to demonstrate that the modification described by this evaluation to the Drywell Continuous Airborne Radioactivity Monitoring (CAM) System within the Reactor Building did not affect overall plant safety. In addition, the modification of the CAM System complies with the overall system design criteria contained in the FSAR.

The purpose of this modification was to improve CAM System reliability and reduce maintenance by replacing the 1 inch (1"-N-1504-10F) sample inlet line associated with CAM 04-1. The sample inlet line was replaced and relocated so that the inlet is near the discharge of the Drywell Coolers. This relocation provided a cooler and drier air sample to the CAM.

This modification of the Drywell Continuous Airborne Radioactivity Monitoring (CAM) System, Did not affect any safety related or environmentally qualified components or systems, or affect overall plant safety. In addition, the modification complies with the overall safety design bases contained in the FSAR. Design and installation complied with relevant codes, standards and criteria.

Based on the review performed herein, the following is concluded for this modification:

The original design basis as stated in JAFNPP FSAR Section 7.12.6 has not been changed.

The proposed modification was evaluated for postulated equipment failures and no new type of scenario exists. The modification was designed and installed in accordance with appropriate codes, standards, and plant procedures that have provided a level of safety and personnel protection that is consistent with the CAM system design basis.

This modification did not affect the Technical Specifications or the Safety Analysis of the plant as described in the JAFNPP FSAR Chapters 14 and 15.

No additional fire load was added to the Reactor Building by the installation of this modification. Post installation checks (e.g., leak and operability tests) ensured the level of quality of the CAM System was not degraded. The Security Plan was unaffected.

Potential discharge to the environment was not affected by this modification. Therefore this modification did not alter the environmental impact of the plant.

In conclusion, the implementation of this modification did not conflict with the design basis of the CAM System as

MODIFICATION: M1-87-050

JAF-SE-91-058 THERMOCOUPLES AND RTD ASSEMBLIES FOR RX VESSEL HEAD

Modify the reactor vessel head thermocouples so that they can be quickly and conveniently assembled/disassembled during refueling operations. The instrumentation channels currently penetrate the drywell inner refueling seal ring and have small threaded connectors which have to be manipulated during refueling operations in plastic suit and respirator.

The thermocouple connectors are located in five inch conduit sleeves mounted on top of the seal ring. The instrumentation penetrations through the seal ring have begun to leak while the vessel head cavity is flooded up due to corrosion. This modification sealed the existing instrumentation penetrations and route the instrumentation channels through existing ventilation hatches in the seal ring. Quick disconnects were installed at the insulation carrousel.

The field wires to the 16-1RTD-110 & 111 sensors had to be determinated prior to disassembly and removal during refueling operations in plastic suit and respirator. Disassembly and removal of these sensors takes time and the radiation dose rate is high. The modified configuration allows the sensors to be unclamped from the insulation carrousel ribbing and passed down through a ventilation hatch into the drywell without any further disassembly. The design configuration of the instrumentation loops has remain unchanged.

The 16-1RTD-110 & 111 loop drawings did not match the existing configuration in the drywell. These loops were as-built under this thermocouple modification.

The reactor vessel instrumentation system is described in the FSAR section 7.8. The reactor coolant system limiting conditions for operation and surveillance requirements are described in Technical Specification section 3.6. The Technical Specifications did not require revision. Table 7.8-1 of the FSAR required a revision to update the "Accuracy" column, the accuracy of the thermocouples was incorrectly listed. The existing and new thermocouple accuracies are equivalent and equal to $\pm$ 1°C or 1.8°F.

The purpose and function of the affected instrumentation channels were not changed by this modification, and therefore, the operation of the Nuclear Boiler Instrumentation (02-3) and PCILRT (16-1) systems were not affected.

This MMP did not constitute an unreviewed safety question pursuant to 10CFR50.59.

MODIFICATION: F1-91-032

JAF-SE-91-067, Rev. 0 POST ACCIDENT HYDROGEN MONITORING SYSTEM - REDUNDANT TORUS SAMPLING LINE REROUTE

A walkdown performed by NYPA during the 1990 refueling outage revealed that the containment H_2/O_2 "B" side Torus sample line was terminated in the ring header instead of the Torus air space. The atmosphere in the ring header is representative of the Drywell atmosphere and not the Torus atmosphere. Therefore, in order to meet the single failure criterion of Reg. Guide 1.97, Rev. 2, Appendix B of NUREG-0737, and the information of NYPA letter JPN-83-38 which describes a commitment confirmed in the NRC's Order of March 14, 1983 (for NUREG-0737), that the existing line will be modified and a redundant Torus sample line will be installed for monitoring the hydrogen concentration in the Torus air space.

The scope of this modification was to correct this condition by disconnecting the existing Drywell torus level sample line from Drywell Penetration 16X-26A and replacing it with a new line. The new line runs from Torus Penetration 16X-216. Existing automatic isolation valves 27SOV-119F1 and -119F2 were removed and relocated near Torus Penetration 16X-216. Existing ILRT valves 27CAD-713, -714, -715 and manual isolation valve 27CAD-205 were removed and discarded. New ILRT valves and a manual isolation valve were installed.

This modification only changed the scheme used for monitoring hydrogen concentration in the torus and all the system components remain the same functionally.

Existing design criteria were implemented for the purchase and installation of the system components.

This modification adhered to all applicable codes and standards for purchasing and installation of the system components.

This modification installed a redundant torus sample line.

Did not cause any alterations to the FSAR accident analysis because installation of this modification did not alter any accident analysis currently addressed in the FSAR.

MODIFICATION: M1-91-198

JAF-SE-91-082, Rev. 2 REPAIR AND UPGRADE OF APPENDIX "R" FIRE DAMPERS

The 1991 draft Fire Protection Reference Manual Section 4.17 which was prepared by ABB Impell Corporation listed 11 recommendations, including the inspection of all dampers passing through fire barriers. NYPA inspections of all plant dampers produced twenty-six fire dampers that did not allow room for thermal expansion and therefore did not comply with the manufacturer's and the UL approved method of installation. All dampers were replaced with 3-hour dampers. It was determined that two additional dampers needed to be upgraded from 1-1/2 hour Appendix A dampers to a 3-hour Appendix R damper (72FD-5) and a 3-hour Appendix A damper (67FD-17).

The twenty-eight fire dampers comprise a portion of a 10CFR50, Appendix R or a BTP 9.5-1 Appendix A fire area boundary. All of the modifications were performed installing 3-hour dampers to insure that the barriers met or exceeded their original rating.

These modifications affected various ventilation systems. Installation/testing of damper 72FD-5 caused both battery room ventilation systems to become inoperable which in-oped the station batteries. Therefore, the plant was required to be in a cold condition to perform this work.

Testing of dampers 67FD-4, 5, 13 resulted in loss of cable tunnel ventilation. This will not affect the operability of the EDG's and the equipment whose cables are routed through the cable tunnel Reference Stone and Webster's calculation number 02268.5004-US(N)-001-0.

The Fire Protection Reference Manual (FPRM) was revised to upgrade the barrier between the Battery Room corridor and the Battery Charging Rooms from an Appendix A barrier to Appendix R, therefore damper 72FD-5 was upgraded to a 3-hour barrier. This modification has already been analyzed by FPRM and had no effect on the FSAR or Technical Specifications. All dampers in safety related systems were seismically qualified to insure proper operation and premature activation should an accident occur.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-91-205

JAF-SE-91-086, Rev. 0 FIRE PROTECTION - BATTERY ROOM CORRIDOR BR5 - AUTOMATIC FIRE SUPPRESSION SYSTEM

This modification provided automatic water fire suppression capability in the Battery Room Corridor (Fire Zone BR-5). The Fire Suppression system was to comply with 10CFR50, Appendix R, Section III.G.3 which required detection and fixed suppression in areas for which alternate shutdown capability is provided. These requirements were identified in the JAFNPP Fire Hazards Analysis as revised in October 1985 and recommended in the 1991 updated Fire Protection Reference Manual, Section 4.17.

10CFR50, Appendix R required fire detection and fixed suppression be provided in areas for which alternate shutdown capability is provided. This modification augmented the Battery Room Corridor Fire Protection system by adding automatic sprinklers in accordance with NFPA 13. This modification resolved licensing concerns of loss of remote shutdown capability due to a fire in Fire Zone BR-5.

Alternative shutdown capability was required for this area as a result of possible loss of Divisions "A" and "B" cabling due to a fire in the area.

The New York Power Authority requested an exemption from the requirements of 10CFR, Part 50, Appendix R, Section III.G.e, which requires NPP Licensees to have an Automatic Fire Suppression System for areas containing redundant trains of safe shutdown cables. On September 18, 1991, NYPA received the exemption since alternate capability is provided independent of the fire area of concern. Any fire or sprinkler actuation that was to occur in the Battery Room Corridor would not affect the ability of the plant to achieve and maintain safe shutdown. Modifications previously implemented in Fire Zone BR-4 to ensure alternative shutdown capability for a fire in the Main Control Room also ensure alternative shutdown capability for a fire or loss of equipment in Fire Zone BR-5. This alternative shutdown capability is provided by the provision of Division "B" dc power to the remote shutdown panels via distribution panel 71DC-B4 (Fire Zone EG-6). The redundant Battery Room Ventilation Control Panels 72HV-5A and 5B are physically separated in the Battery Room Corridor. The sprinkler heads are independently activated by heat only and located recessed above the structural beams preventing them from being damaged. If one sprinkler head is activated, it will not activate another head and the spray from one sprinkler will not splash on both panels at the same time. If ventilation is lost, temporary ventilation can be established by using plant Procedure AOP-58.

JAF-SE-91-086, Rev. 0 **FIRE PROTECTION - BATTERY ROOM CORRIDOR BR5 - AUTOMATIC FIRE SUPPRESSION SYSTEM** (continued)

This will preclude loss of both systems if a break in a line is isolated. In addition, there are hose stations in the Turbine Building and Administration Buildings that are capable of reaching the Battery Room Corridor.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-90-212

JAF-SE-91-096, Rev. 1 UPGRADE OF EPIC HARDWARE AND SOFTWARE

The total project work scope included modifying the existing EPIC software to operate under the latest version of VMS and combining the processes that were currently running on two VAX 11/785s to run on one VAX 6510. Both software and hardware modifications were designed to provide for increased performance of the EPIC system allowing for expansion of available functions.

This modification to the EPIC system replaced the EPIC computing hardware and upgrade the existing software. The new software provided for system database expansion and increased efficiency. The operation of the EPIC system including operational displays were not changed by this modification.

Based on the original safety reviews performed for the EPIC system and a specific review of the software and hardware changes made within the EPIC system, the following was concluded for the Safety Evaluation.

The modification did not affect the functional operation, safety systems, or the instrumentation of the plant.

The modification did not affect the functional operation, safety systems, or the instrumentation of the plant. The new software and hardware was purchased to specifications that met or exceeded the original specification requirements.

This modification did not affect the Technical Specifications or the safety analysis of the plant as described in the JAFNPP FSAR Chapter 14.

This modification did not affect the Fire Protection, it did not impact any safe shutdown components of fire barriers, and it did not increase the combustible levels in any of the plant areas.

This modification only concerns the handling and processing of plant data.

MODIFICATION: M1-91-039

JAF-SE-91-100 INSTALLATION OF R.G. 1.97 FUSES FOR 71L-15 AND 71L-16

The purpose of this modification was to provide fuse protection between Regulatory Guide 1.97 instruments, 71VM-L15-2 and 71VM-L16-2 (located in panel 09-8), and the local voltmeter (71VM-L15-1 and 71VM-L16-1) and undervoltage relay circuits located in 71L-15-1A and 71L-16-1A. The fuse protection was required due to the local voltmeters and undervoltage relays not being environmentally qualified. In their previous configuration, failure of 71VM-L15-1 and 71VM-L16-1 could result in the inoperability and termination of voltage monitoring of emergency busses 11500 and 11600 in the Control Room due to interconnecting wiring. This modification installed two (2) five amp fuses, one (1) fuseholder and revised the existing wiring of the undervoltage relay in each cubicle (71L-15-1A and 71L-16-1A). The fuses were installed in series with the voltmeter, undervoltage relay and voltmeter switch. This configuration provided sufficient protection and isolation for the R.G. 1.97 instrumentation in the Control Room in the event of local voltmeter or undervoltage relay failure.

The AC Distribution System is described in FSAR Section 8.5 and has a power generation design basis as outlined in FSAR Section 8.5.1. The AC Distribution System also has a safety design bases as outlined in FSAR Section 8.5.3. This modification complied with both FSAR Sections 8.5.1 and 8.5.3 and also did not affect or change Sections 3.2 and 3.9 of the Technical Specifications. This modification did not increase the probability or consequences of an accident postulated in Chapter 14 of the FSAR and did not affect the Fire Protection System as put forth in Chapter 9.0 of the FSAR.

This modification resolved the Regulatory Guide 1.97 issue regarding interaction between Regulatory Guide 1.97 and Non-Regulatory Guide 1.97 components in a common circuit. In the event of a local voltmeter or undervoltage relay failure, the isolation fuses will open, Control Room indication on 71VM-L15-2 and 71VM-L16-2 for bus 11500 and 11600 will remain unaffected and voltage monitoring will be maintained. The design and function of the L15 and L16 switchgear have not been changed or degraded by this modification.

This design change does not constitute an unreviewed safety question pursuant to 10CFR50.59.

MODIFICATION: N/A

JAF-SE-91-105, Rev. 0 CHEMICAL DECONTAMINATION OF REACTOR WATER RECIRCULATION SYSTEM AND PORTIONS OF THE REACTOR PRESSURE VESSEL, RESIDUAL HEAT REMOVAL AND REACTOR WATER CLEAN-UP SYSTEMS

Due to increasing radiation levels on the out-of-core piping and the resultant higher occupational radiation exposures for outage jobs conducted in the drywell, there was a need to lower the radiation levels at JAF. One of the major contributors to drywell exposure is the oxide layer inside the RWR piping and adjacent systems that contain activated corrosion products. Chemical decontamination by dilute solvents is a widely accepted method to remove this oxide layer without detrimental effects on the components and piping systems that will be returned to service. The solvent systems chosen were based on the experience gained during the chemical decontamination at JAF performed in September 1988 and subsequent recontamination information gathered since that decontamination. The cross-tie solvent was changed to a regenerative process based on the JAF and industry experience with carbon steel systems. The oxidation process was changed from NP to AP due to the lower corrosion rates associated with the AP chemistry.

This Safety Evaluation addresses the chemical decontamination of the James A. FitzPatrick (JAF):

- Reactor Water Recirculation (RWR) System
- Portions of the Reactor Water Cleanup System (RWCU)
- Portions of the Residual Heat Removal System (RHR)
- The RHR to Fuel Pool Cooling (FPC) cross-tie piping
- The jet pump instrumentation (JPI) drain line
- The recirculation sample line
- The Crack Arrest Verification (CAV) system
- The reactor vessel annulus region below the slip joints of the jet pumps.

It was determined from this review that the chemical decontamination of the RWR system and portions of the RHR/FPC cross-tie line resulted in the following:

The operation of the systems were not altered.

Exposure to the decontamination solvent causes no structural integrity concerns or impact on system operations for those components that were returned to service.

System operation was not altered by chemical decontamination of these components.

JAF-SE-91-105, Rev. 0 CHEMICAL DECONTAMINATION OF REACTOR WATER RECIRCULATION SYSTEM AND PORTIONS OF THE REACTOR PRESSURE VESSEL, RESIDUAL HEAT REMOVAL AND REACTOR WATER CLEAN-UP SYSTEMS
(continued)

No additional fire load was being added to the facility because the chemicals and the resins were not explosive.

The potential discharge of liquid radioactive effluents to the environment were not affected. Solid waste was removed to approved disposal sites by the Radioactive Waste disposal vendor.

MODIFICATION: M1-90-218

JAF-SE-91-107, Rev. 0 HPCI OIL FILTER VENTING SYSTEM MODIFIED

The Maintenance Department requested the HPCI Turbine Oil Filters 23TOF be modified to install a petcock to bleed air after changing the oil filters because the present method to bleed air from the HPCI Turbine Oil System presents a potential personnel safety hazard. Venting is required per MP101.11 steps 7.2.15 and 7.3.12. This modification allows better control in performing this procedure. This modification reduced radiation exposure by simplifying the filter bleeding process.

The HPCI turbine oil filter is QA Cat. I and Seismic Class I per the FSAR Section 16.4 and MEL. The addition of air bleed valves on the HPCI turbine oil filters did not affect the HPCI system function and operability.

The new air bleed valves provides better control in removing air from the filters under system pressure following filter cartridge replacement. The installation of air bleed valves in the HPCI turbine oil filters did not increase the probability or consequence of an accident or malfunction analyzed in the FSAR nor did it create the possibility of an accident or malfunction not previously analyzed. It did not reduce the safety margin established by the Technical Specification.

This proposed modification did not constitute an unreviewed safety question pursuant to 10CFR50.59.

MODIFICATION: M1-89-048

JAF-SE-91-108, Rev. 0 HPCI TURBINE EXHAUST DRAIN POT LEVEL SWITCH REPLACEMENT

This modification replaced the HPCI Turbine Exhaust Drain Pot Level Switch (23LS-98). This existing Robertshaw switch had a relatively narrow operating range (one half inch) before the switch activates to empty the associated drain pot. Also the present horizontal vessel configuration traps dirt that fouls the switch linkage. This switch had a history of work requests including at least four occurrences of false switch operation since 1986.

This modification replaced the HPCI Turbine Exhaust Line Drain Pot level switch which is QA Category II/III as stated in the Master Equipment List and the System Flow Diagram.

The replacement HPCI Turbine Exhaust line level switch met applicable design and safety basis requirements. The replacement switch improved the reliability of the HPCI exhaust line drain pot. The level switch did not perform any safety function. A Seismic Class I support was installed to maintain the integrity of the pressure boundary of the HPCI exhaust line drain pot which is safety related.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-91-242

JAF-SE-91-109 RHR PUMP DISCHARGE ORIFICE RESIZE

This proposed modification changed the four RHR pump runout restriction orifices 10RO-107A thru 107D. Smaller orifices were used to reduce the RHR pump discharge flow.

This plant modification to the Residual Heat Removal System pump discharge orifice was classified as safety related, QA Category I, and Seismic Class I.

New orifice plates were being installed to reduce the flow rate of the RHR pumps to a more efficient operating point on the pump curve. The new orifice plates have been sized for 9500 gpm LPCI pump flow at 20 psi reactor pressure, exceeding the minimum 8,910 GPM at 20 psi reactor pressure as required by the FitzPatrick Technical Specification, section 4.5.A.3. This design margin for sizing the orifice plates were sufficient to (1) account for possible instrument inaccuracies for the instruments that will be used to measure pump flow during the pre-operational test (POT-10T) and subsequent surveillance testing (ST-2A). (2) to account for possible pump degradation based on the acceptance ranges for pump performance that are provided in the Surveillance test (ST-2A) which are based on current Technical Specifications.

The approximately 5% reduction of flow in the various modes of RHR has no impact on the conditions that would initiate an accident. The consequences of an accident would not be affected since the flow rates will remain higher than that considered in the analyses which support current Technical specification flow rate requirements.

The orifices did not effect the actual response of the RHR system to abnormal or accident conditions nor can it produce a new accident not previously analyzed in the FSAR.

The new orifice was designed to meet the current LPCI flow requirements outlined in the Technical Specifications of 8910 gpm against 20 psid between the reactor vessel and the suppression pool with a flow design margin of approximately 590 gpm. The previous flow design margin was 400 to 900 gpm above the accepted Technical specification LPCI flow requirement of 9900 gpm against 20 psid between the reactor vessel and the suppression pool. The design flow rate decreased by about 5%, but this did not impact the JAFNPP licensing basis and LPCI remains capable of performing its intended function during postulated LOCAs.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-90-063

JAF-SE-91-115 FUSE REPLACEMENT IN 125VDC DISTRIBUTION SYSTEM

The purpose of this modification was to resolve one part of unresolved item no. 89-80-10 in the NRC Safety System Functional Inspection Report. Specifically, the modification resolved the coordination between the 35 ampere fuse and the 40 ampere circuit breaker in the 125VDC control circuits of the 4160 volt air circuit breakers (ACB's) and the 30 ampere fuse and the 40 ampere circuit breaker in the 125VDC control circuits of the 600 volt ACB's.

The modification consisted of replacing existing trip circuit fuses in all type AMH and type AKD circuit breakers with 20 ampere fuses in order to achieve coordination between trip circuit fuses and their 40 ampere feeder breaker. The maximum trip circuit load is less than 9 amperes. This load can safely be supplied by a 20 ampere fuse.

The fuses and fuse holders were procured as QA Cat. I (safety-related) items and were environmentally qualified to the requirements of the JAF EQ program.

The safety evaluation concluded that the modification did not involve a change in the Technical Specifications or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-91-081

JAF-SE-91-117, Rev. 1 SUPPRESSION POOL TEMPERATURE ALARM SETPOINT EVALUATION

The purpose of this modification was to reprogram the deviation alarms, and restore the 16th RTD in bay "G" for the suppression Pool Bulk Temperature Monitoring System (SPBTMS).

The design intent of SPBTMS was to measure individual bay temperature (total of 16), calculated the bulk (average) temperature, and display both bulk (on 09-3 Panel) and individual temperature, and display both bulk (on 09-3 Panel) and individual (on 27MAP Panel) bay temperature. The plant operator will follow prescribed procedure to initiate necessary action to monitor Suppression Pool Bulk Temperature in response to any alarms.

The changes in the FSAR are for the installation of the 16th RTD and removal of jumpers installed under Temporary Mod. No. 90-106, and did not affect margin of safety as evaluated in the nuclear safety analysis report.

This MMP did not constitute an unreviewed safety question pursuant to 10CFR50.59.

MODIFICATION: M1-88-073

JAF-SE-91-119, Rev. 0 REPLACEMENT OF CONDENSER DRAIN MANIFOLD

The purpose of this modification was to provide a replacement drain manifold for line number 41-12"-SDD-602-54. The manifold had experienced severe wall thinning due to erosion/corrosion. The new manifold (41-12"-SDD-603-54) was manufactured using material which has greater erosion/corrosion resistant properties than the original material used.

The drain manifold which was being replaced is part of system 41 (Miscellaneous Drains, Secondary Plant) which is defined as a QA Category II/III system per section 12.2 of the FSAR.

The replacement manifold satisfied all of the design requirements for the application and has restored the structural integrity of this portion of the drain piping system. This modification did not affect the operation of any safety-related systems or components, did not involve a change to any JAF Technical Specification and did not involve an unreviewed safety question.

MODIFICATION: M1-91-122

JAF-SE-91-126, Rev. 0 EAST CRESCENT TEMPERATURE SENSORS ACCESS PLATFORMS

Previously, scaffolding had to be installed in the east crescent above the HPCI pump to obtain access to the four temperature sensors located on the east crescent ceiling in order to perform Surveillance Testing in accordance with ST-76J45. Long term installation of extensive scaffolding was undesirable in safety-related areas. To eliminate the use of the temporary scaffolding for each surveillance testing and consequently reduce worker radiation exposure, this modification provided the fabrication and installation of two permanent platforms that were built on existing building steel.

The new platforms are QA Category II/III and seismic class II. The platforms are constructed of miscellaneous structural steel and they perform no safety function. An engineering calculation has been completed and has determined that the platforms are adequately designed for worst case loading conditions involving deadweight and seismic loading. This modification eliminated the need for scaffolding installation to perform surveillance testing.

This modification did not adversely affect any safety-related equipment in the plant nor did it affect the JAF Technical Specifications. This modification did not affect the JAF FSAR. Based on this evaluation, this modification did not constitute an unreviewed safety question pursuant to 10CFR50.59.

MODIFICATION: M1-91-069 TEST: POT-10R

JAF-SE-91-127, Rev. 0 REPLACEMENT OF 10A-K93A & 10A-K93B TIME DELAY RELAYS IN RHR/RCIC RELAY PANELS (09-32/33)

Modification M1-91-069 replaced existing General Electric (GE) CR2820B time delay relays with Agastat ETR series. The replacement relays were installed in the Residual Heat Removal (RHR) logic circuits. A preoperational test was necessary to verify that the replacement relays time delay function (180 seconds) and logic circuits were not altered. The relays ensure that RHR Heat Exchanger Bypass Valves 10MOV-66A and 10MOV-66B remain open after a LOCA signal.

Performance of this preoperational test did not have any adverse affect on plant safety. The test verified the replacement relays time delay function and operability were performed after modification installation activities were completed. Performance of the test had no effect on any other plant system or component. The test was performed on one redundant loop (RHR) at a time and was performed if the conditions of Technical Specification 3.5.F can be met.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specifications or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: TEST: POT-23H

JAF-SE-91-138, Rev. 1 TESTING HPCI PUMP DISCHARGE TO REACTOR INBOARD ISOLATION VALVES UNDER SIMULATED DEGRADED VOLTAGE CONDITIONS

The purpose of the safety evaluation was to document that the performance of pre-operational test procedure, POT-23H did not involve an unreviewed safety question. The scope of the evaluation covered operational tests of valves 23MOV-19 and 23MOV-20 under simulated degraded voltage conditions as defined in NRC GL 89-10. The tests were performed with the plant in a shutdown and cold condition.

There were no statements or descriptions in the FSAR which were in conflict with the performance of this test.

There were no technical specification requirements for availability of the station batteries when the plant is in a cold shutdown condition.

The 125VDC distribution bus B was powered by its associated battery during the test with the battery charger disconnected, but in active standby condition. The bus voltage was maintained above 105 VDC except for brief voltage dips due to actuator motor inrush current at the start of the open and close cycles. An evaluation shows that these voltage conditions did not affect the performance of connected plant loads.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-91-306

JAF-SE-91-139, Rev. 0   NODAL NETWORK COMMUNICATIONS SYSTEM

This modification installed an all digital nodal processor network telecommunication system, the new system provides 100% redundancy for critical voice and data circuits. Critical circuit outages on the new system are automatically detected and re-routing is accomplished without user intervention, reliance on common carriers for re-routing will not be necessary.

The equipment modified has no effect on the licensing basis accidents with radiation consequences detailed in FSAR chapter 14 or fires detailed in FSAR chapter 9. This equipment did not have a safety function. This equipment was not capable of affecting any technical specification parameters.

The purpose and function of the affected telecommunication equipment were not changed by this modification. The performance capabilities of this equipment was improved and is the justification for this modification.

This MMP did not constitute an unreviewed safety question pursuant to 10CFR50.59.

MODIFICATION: M1-88-244

JAF-SE-91-140, Rev. 0 EDG FUEL FILTER REPLACEMENT

The emergency diesel generator (EDG) fuel oil system was modified as follows:

A. Replaced the canister type duplex fuel oil filter assemblies (93EDP-F-A1, B1, C1, D1 and 93P-F-A4, B4, C4, D4) with twin spin-on type duplex fuel oil filter assemblies (93F-5A, 5B, 5C, 5D and 93P-4A, 4B, 4C, 4D) to eliminate fuel leaks and better availability of replacement parts associated with the canister type fuel oil filter assemblies.

B. Installed differential pressure (dp) gauges for both filter assemblies to detect filter plugging. In the past JAF could not detect fuel filter plugging until engine performance is degraded. Without dp gauges, the existing duplex filters were operated in parallel, maximizing the filter capacity while minimizing our potential of filter plugging. The addition of dp gauges allows the machines to be operated in single element operation and when the element in service becomes dirty (based on a differential pressure increase) the control valve on the filter housing may be moved to operate the other filter. This allows the dirty filter to be replaced during EDG operation without performance degradation.

C. Switched the suction lines between the engine (93EDP-A1, B1, C1, D1) and the motor driven fuel pumps (93P-4A, 4B, 4C, 4D). This will lineup the emergency fuel shutoff valve (93EDG-101A, 101B, 101C, 101D) with the engine driven pump instead of the motor driven pump, allowing operators to isolate all fuel delivery to the engine in the event of an emergency by taking two actions. Currently there is no means of isolating all fuel oil to the engine.

This modification did not adversely impact any safety related components or systems, or affect overall plant safety. The design and installation comply with the original codes, standards and criteria. This modification complied with the overall system design criteria contained in the FSAR.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-89-145

JAF-SE-91-141, Rev. 0 REMOVABLE DRYWELL MONORAIL (EAST EQUIPMENT HATCH)

During refueling and maintenance outages numerous large heavy components were lifted in and out of the drywell through the east equipment hatch. The load path involved a vertical and horizontal transfer of only a few feet. However, the low ceiling of the hatch results in having to sling from inside the drywell and drift the load through the hatchway. This modification used a monorail to provide a controlled, safe rigging system that was easy to install and remove for plant outage use.

Implementation of this modification provided a pre-engineered monorail that can be readily installed in the east drywell equipment hatch on an as-needed basis for plant outage use. The design includes a simple cantilever structure permanently anchored on the outside of the drywell wall to facilitate the support of the removable trolley beam. The monorail provides a controlled, safe rigging system to transfer various heavy loads (up to 1 ton) through the drywell equipment hatch.

The trolley beam was permanently marked with the maximum safe working load and the governing installation drawing number. This new plant drawing provides an assembly detail and a representation of the safe load path.

The monorail structure was designed such that member stresses are within the applicable allowable limits, and the maximum safe working load of the individual components were not exceeded. A 10:1 safety factor, typically used in the design of rigging systems for heavy loads (greater than 750 lbs. at JAFNPP), was not required for this modification since a postulated load drop was evaluated in accordance with NUREG-0612 as follows: The monorail will only be installed and available for use during a plant outage when maintenance is to be performed inside the drywell. Directly below the load path for the trolley hoist, there are no components whose failure could result in lost shutdown capabilities nor releases of radioactive material which could exceed off-site dose limits. The effects of an uncontrolled load drop are further minimized by the fact that the load path is limited to only a few feet both vertical and horizontal, and is directly over floor beams and grating in the drywell.

This proposed structural modification did not affect the JAF Technical Specifications and did not constitute an unreviewed safety question pursuant to 10CFR50.59.

MODIFICATION: M1-91-353

JAF-SE-91-143, Rev. 0 FIRE PROTECTION EXTENSION FOR NEW TEMPORARY OFFICE ANNEX

This modification provided fire protection for the new temporary office area located adjacent to the Administration Building Annex. The new temporary office modules, with approximately 2500 square feet, has a fire protection sprinkler system installed that will be supplied from an existing branch header that services the existing Administration Building Annex office areas.

This modification extended and modified an existing 3 inch sprinkler branch header to provide fire protection to new temporary office modules adjacent to the Administration Annex complex. This modification met the requirements of NFPA-13 (Standard for the Installation of Sprinkler Systems). This modification is classified QA Category II/III and did not adversely affect the design bases as described in JAF-FSAR Section 9.8. The sprinkler system in the new temporary office area has no safety related function associated with it and it is not in a safety related area.

JAF Technical Specifications were not affected by this modification because operation of this portion of the plant Fire Protection system was not important to safety. This modification did not adversely affect the Q.A. Category M portion of the plant Fire Protection system. The plant Fire Protection System has sufficient capacity to supply the added sprinklers in the new temporary office modules when compared to the capacity and rating of the existing fire pump.

In conclusion, this modification to the JAF Plant Fire Protection System did not conflict with the objectives and design basis of the Plant Fire Protection System as stated in the FSAR, nor result in changes to the Technical Specifications. Additionally this modification was consistent with the guidance and programs described in BTP 9.5-1 Appendix A for fire protection systems. Safety related structures, systems or components were not affected by this modification.

This modification did not constitute an unreviewed safety question pursuant to 10CFR50.49.

MODIFICATION: M1-91-211

JAF-SE-92-001, Rev. O   INSTALLATION OF ULTRA-VIOLET FLAME DETECTOR

The purpose of M1-91-211 was to install a fire detection system in the Turbine Building (Fire Area 1E) adjacent to the East and West Electric Bays. This modification installed one (1) Ultra-Violet Flame Detector, one (1) Ultra-Violet Flame controller and associated raceway and cable. The Fire Detection System was required because Fire Area 1E was without fire detection or suppression capabilities. Although a technical exemption to 10CFR Part 50, Appendix R, Section III.G.3 has been granted by the Nuclear Regulatory Commission from having fire detection capabilities in fire area 1E (adjacent to the east and west electric bays), NYPA committed to having a fire detection system in place, at this location, as part of the Fire Protection Program.

The Fire Protection System is described in the Final Safety Analysis Report, Section 9.8, and has a design basis as outlined in Section 9.8.2. M1-91-211 complies and provides an addition to Section 9.8.3 of the FSAR and does not affect or change Section 3.12 of the Technical specifications. M1-91-211 did not increase the probability or consequences of an accident postulated in Chapter 14 of the FSAR.

M1-91-211 resolved the 10CFR Part 50 Appendix R issue pertaining to the requirement to have fire detection in the Turbine Building (Fire Area 1E). Utilizing the Ultra-Violet flame detector to monitor the area where 67HV-2A and 67HV-2B are located provides a safe and reliable means of fire detection. M1-91-211 did not affect or change the design and function of the existing Fire Protection system.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-82-017

JAF-SE-92-004, Rev. 1 B-LOOP CORE SPRAY AND SAFE-END REPLACEMENT

The purpose of this modification was to replace the Core Spray B Loop stainless steel safe-end, associated thermal sleeve and piping out to the first manual isolation valve (14CSP-14B). This modification cut and removed the original TP 304 stainless steel safe-end and piping out to the isolation valve 14CSP-14B, and welded on a replacement safe-end and piping, manufactured and fabricated from TP 347 modified stainless steel material.

The FSAR and Technical Specifications have been reviewed and in no case does the implementation of this modification create an unreviewed safety question pursuant to 10CF50.59 (References V. 3&4), nor does it increase the probability of a previously evaluated accident.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: Temp Mod 92-014

JAF-SE-92-005 ALTERNATE CO_2 SUPPLY FOR MAIN GENERATOR

The purpose of Temporary Modification 92-014 was to provide an alternate supply of CO_2 to the Main Generator Hydrogen Cooling System for purging the generator of hydrogen. This temporary modification was required because the normal CO_2 supply vaporizer was out of service. This safety evaluation evaluates the use of the alternate CO_2 supply, which constitutes the "operation of a system different than that described in the JAF FSAR" (FSAR Section 9.8.3.2).

The alternate CO_2 source did not create the possibility of an accident or malfunction of a different type than any described in the FSAR.

The CO_2 purge system and Main Generator Hydrogen Cooling System are not discussed in the basis for any Technical Specifications. This temporary modification did not reduce the margin of safety as described in the basis for any technical specification.

The safety evaluation concluded that the temporary modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION:

JAF-SE-92-006, Rev. 1   CHEMICAL TREATMENT OF SCREENWELL FOREBAYS TO REMOVE ZEBRA MUSSELS

The purpose of this treatment program is to remove zebra mussels from the screenwell and circulating and service water systems by adding a molluscicide (Betz Clamtrol CT-1) to the screenwell and circulating it with RHR service water (RHRSW), normal service water, fire protection water, and circulating water pumps. To provide effective chemical at least 60°F. As this was done during the 1992 refueling outage, reactor and fuel pool decay heat and pump heat was used to raise the water temperature. This safety evaluation addressed the effects of this chemical treatment procedure on safety related equipment and systems and the availability of the ultimate heat sink during and after the completion of the chemical treatment process.

The following items were reviewed for their potential effect on the safety related equipment and systems and their proper operation:

1) Operation of safety related equipment during the heat up and chemical treatment process.

2) Corrosive effects of the chemicals to be circulated on safety related and non-safety related equipment.

3) Availability of the ultimate heat sink during chemical treatment or during loss of offsite power.

The safety evaluation concluded that the Chemical Treatment did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-91-037

JAF-SE-92-008 INSTALLATION OF NEW SWS SUPPLY HEADER ISOLATION VALVES (46SWS-218 A & B)

The purpose of this modification was to provide for the installation of new isolation valves (46SWS-218A and B) downstream of check valves 46SWS-60A and B. The new isolation valves give plant personnel the ability to perform maintenance activities on check valves 46SWS-60A and B during plant operation. Maintenance work on check valves 46SWS-60A or B required the isolation of service water to all of the crescent area unit coolers which would require the removal of crescent area unit coolers 66UC-22(A-K) from service. However, the installation of isolation valves downstream of 46SWS-60A and B gave plant personnel the ability to supply cooling water to the crescent area unit coolers by using the emergency service water system while working on check valves 46SWS-60A or B.

The locations of the new isolation valves (just downstream of 46SWS-60A and B check valves) corresponds to the recommended location provided by the emergency service water system/service water system action plan (reference 12.7).

There were no changes to the design functions of the affected normal and emergency service water piping systems as a result of the installation of the new isolation valves. This modification did not affect the Technical Specifications and did not constitute an unreviewed safety question pursuant to 10CFR50.59.

MODIFICATION: M1-87-094

JAF-SE-92-009 **ASCO SOLENOID VALVE REPLACEMENT FOR SDIV ISOLATION TEST SOLENOID VALVE AND FEED WATER HEATER EXTRACTION STEAM SUPPLY 31NRV-115A SOLENOID VALVES**

This modification evaluated replacement of two models of ASCO solenoid valves. For 31SOV-115A&B, -116A&B, these valves were no longer manufactured and this modification evaluated a replacement valve. For 03SOV-29, the valve had too large a flow coefficient (C_v), causing valve testing times for scram discharge vent and drain AOVs to be undesirable fast as compared to actual (RPS initiated) closing times.

JAF modification M1-87-094 replaced 03SOV-29, the SDIV air header test solenoid, with a solenoid valve with a reduced flow coefficient. This was required to allow the test solenoid to more closely duplicate SDIV vent and drain valve closure times obtained via testing with the RPS solenoid valves, 03SOV-31A,B. (Reference 12.4)

Additionally, modification M1-87-094 replaced 31SOV-115A,B; -116 with equivalent solenoid valves manufactured by the same vendor, ASCO, due to discontinuation of the original solenoid valve mode.

The components affected by this modification are QA Category II/III.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: **TEST: STP-10U**

JAF-SE-92-013 **IN-SITU DESIGN BASIS DIFFERENTIAL PRESSURE TEST FOR RHR TORUS COOLING ISOLATION VALVES**

Special Test STP-10U was performed to address requirements from NRC Generic Letter 89-10 which dictate in-situ design basis testing of motor operated valves (MOVs). This evaluation covers testing of the following Residual Heat Removal (RHR), System 10, valves.

- 10MOV-34A RHR A Torus Cooling Throttle Valve
- 10MOV-34B RHR B Torus Cooling Throttle Valve
- 10MOV-39A RHR A Torus Cooling Isolation Valve
- 10MOV-39B RHR B Torus Cooling Isolation Valve

The test established simulated design basis differential pressure across each valve to be tested, and then directed the stroking of the valve. Valve differential pressure was obtained from a combination of temporary and permanently installed instrumentation. Valve operator characteristics was monitored and recorded using Liberty Technology's "VOTES" test equipment.

This Nuclear Safety Evaluation allows performance of differential pressure testing with flow of STP-10U. The purpose of STP-10U is to simulate design basis differential pressure conditions across valves 10MOV-34A (B) and 10MOV-39A (B) during stroking of these valves. Valve operator characteristics were monitored and recorded using Liberty Technology's "VOTES" test equipment. STP-10U may be performed in any reactor operational mode when all required Technical Specifications are met.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A STP-10V

JAF-SE-92-014, Rev. 0 IN-SITU DESIGN BASIS DIFFERENTIAL PRESSURE TEST FOR RHR HEAT EXCHANGER BYPASS VALVES AND RHR TO RVS SPRAY ISOLATION VALVES

Special Test STP-10V was performed to address requirements from NRC Generic Letter 89-10 which dictate in-situ design basis testing of motor operated valves (MOVs). This evaluation covers testing of the following Residual Heat Removal (RHR), System 10, valves.

- 10MOV-38A RHR A Torus Spray Isolation Valve
- 10MOV-38B RHR B Torus Spray Isolation Valve
- 10MOV-66A RHR A Heat Exchanger Bypass Valve
- 10MOV-66B RHR B Heat Exchanger Bypass Valve

The test established simulated design basis differential pressure across each valve tested, and then directed the stroking of the valve. Valve differential pressure was obtained from a combination of temporary and permanently installed instrumentation. Valve operator characteristics were monitored and recorded using Liberty Technology's "VOTES" test equipment.

This Nuclear Safety Evaluation allowed performance of differential pressure testing with flow of STP-10V. The purpose of STP-10V is to simulate design basis differential pressure conditions across valves 10MOV-38A (B) and 10MOV-38A (B) during stroking of these valves. Valve operator characteristics were monitored and recorded using Liberty Technology's "VOTES" test equipment. STP-10V may be performed in any reactor Cold Condition only.
The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A **STP-10X**

JAF-SE-92-016, Rev. 0 **IN-SITU DESIGN BASIS DIFFERENTIAL PRESSURE TEST FOR RHR SUCTION-TORUS ISOLATION VALVES**

Special Test STP-10X was performed to address requirements from NRC Generic Letter 89-10 which mandates in-situ design basis testing of motor operated valves (MOVs). This evaluation covers testing of the following Residual Heat Removal (RHR), System 10, valves.

- 10MOV-13A RHR Pump A Suction - Torus Isolation Valve
- 10MOV-13B RHR Pump B Suction - Torus Isolation Valve
- 10MOV-13C RHR Pump C Suction - Torus Isolation Valve
- 10MOV-13D RHR Pump D Suction - Torus Isolation Valve

The test established simulated design basis differential pressure across each valve tested, and then directed the stroking of the valve. Valve differential pressure was obtained from a combination of temporary and permanently installed instrumentation. Valve operator characteristics were monitored and recorded using Liberty Technology's "VOTES" test equipment.

This Nuclear Safety Evaluation allows performance of static differential pressure testing by STP-10X. The purpose of STP-10X is to simulate design basis differential pressure conditions across valves 10MOV-13A (B, C, D) during stroking of these valves. Valve operator characteristics were monitored and recorded using Liberty Technology's "VOTES" test equipment.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A STP-10Y

JAF-SE-92-017, Rev. 0 IN-SITU DESIGN BASIS DIFFERENTIAL PRESSURE TEST FOR BOTH RHR CONTAINMENT SPRAY INBOARD AND OUTBOARD ISOLATION VALVES

Special Test STP-10Y was performed to address requirements from NRC Generic Letter 89-10 which mandates in-situ design basis testing of motor operated valves (MOVs). This evaluation covers testing of the following Residual Heat Removal (RHR), System 10, valves.

- 10MOV-26A RHR Containment Spray Outboard Isolation Valve
- 10MOV-26B RHR Containment Spray Outboard Isolation Valve
- 10MOV-31A RHR Containment Spray Inboard Isolation Valve
- 10MOV-31B RHR Containment Spray Inboard Isolation Valve

The test established design basis static differential pressure across each valve tested, and then directed the stroking of the valve. Valve differential pressure was obtained from a combination of temporary and permanently installed instrumentation. Valve operator characteristics were monitored and recorded using Liberty Technology's "VOTES" test equipment.

This Nuclear Safety Evaluation allows performance of static differential pressure testing by STP-10Y. The purpose of STP-10Y was to simulate static design basis differential pressure conditions across valves 10MOV-26A (B) and 10MOV-31A (B) during stroking of these valves. Valve operator characteristics were monitored and recorded using Liberty Technology's "VOTES" test equipment. STP-10Y may be performed only during reactor Cold Conditions.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A TEST: STP-14C

JAF-SE-92-020, Rev. 0 IN-SITU DESIGN BASIS DIFFERENTIAL PRESSURE TEST FOR BOTH CORESPRAY DISCHARGE MINIMUM FLOW VALVES AND CORESPRAY FULL FLOW TEST ISOLATION VALVES.

Special Test STP-14C was performed to address requirements from NRC Generic Letter 89-10 which mandates in-situ design basis testing of motor operated valves (MOVs). This evaluation covers testing of the following Core Spray (CS), System 14, valves.

- 14MOV-5A CS 14P-1A Discharge Minimum Flow Valve
- 14MOV-5B CS 14P-1B Discharge Minimum Flow Valve
- 14MOV-26A CS System A Full Flow Test Isolation Valve
- 14MOV-26B CS System B Full Flow Test Isolation Valve

The purpose of STP-14C was to simulate design basis differential pressure and flow conditions across valves 14MOV-26A (B) and 14MOV-5A (B) during stroking of these valves. Valve differential pressure and flow was obtained from a combination of temporary and permanent plant instrumentation. Valve operator characteristics were monitored and recorded using Liberty Technology's "VOTES" test equipment.

This Nuclear Safety Evaluation allows performance of differential pressure testing under flow conditions by STP-14C for valves 14MOV-26A (B) and 14MOV-5A (B). STP-14C will be performed under similar Core Spray Functional Testing conditions currently allowed by ST-3A and ST-3P, References 5.3, 5.4. STP-14C may be performed in any reactor operational mode when all required Technical Specifications are met.

The safety evaluation concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A TEST: STP-46E

JAF-SE-92-027 IN-SITU DESIGN BASIS DIFFERENTIAL PRESSURE TEST FOR BOTH RBC SUPPLY EMERGENCY SERVICE WATER INJECTION VALVES AND EMERGENCY SERVICE WATER PUMP TEST VALVES.

Special Test STP-46E was performed to address requirements from NRC Generic Letter 89-10 (Reference V.7.4) which mandates in-situ design basis testing of motor operated valves (MOVs). This evaluation covers testing of 46MOV-101A (B) RBC Supply Emergency Service Water A (B) Injection Valve and 46MOV-102A (B) Emergency Service Water Pump A (B) Test Valve.

The test established design basis differential pressure across each valve, and then directed the stroking of the valve. Valve differential pressure and flow characteristics were obtained and compared to valve performance. Valve operator characteristics were monitored and recorded using Liberty Technology's "VOTES" test equipment.

The information collected during this test will be used to continue MOV analysis and trending in accordance with the James A FitzPatrick Generic Letter 89-10 Program Plan.

The safety evaluation determined that tne test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A

JAF-SE-92-030, Rev. 1 ACCEPTABILITY OF ATTS SCRAM SIGNAL PROPAGATION DELAYS DUE TO A TRANSIENT SUPPRESSION DIODE

The purpose of this evaluation was to determine the acceptability of a 42 millisecond (msec) increase in relay drop out time for the Reactor Protection system (RPS), Analog Transmitter/Trip System (ATTS) control relays. This evaluation shall apply to all RPS scram channels which are processed by the ATTS system.

The addition of transient suppression diodes to the ATTS control relay coils increases the response time of the ATTS sensor channel from 210 msec to 252 msec. This value is well below the maximum allowable analyzed limit of 500 msec required by JAF licensing bases (Reference Reactor Protection System Specification (GE Dwg. No. 22A3083AJ, JAF Dwg. No. 16.05-7C. This change is therefore acceptable.

The safety evaluation concluded that the addition of transient suppression diodes to the ATTS control relay coils did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A

JAF-SE-92-032, Rev. 0 END OF CYCLE 10 FUEL INSPECTION

This work was done to achieve the NYPA goal of high fuel reliability. Certain inspections identify specific leaking fuel rods in bundles designated to be failed by vacuum sipping. Other inspections measure the effects of operation on channels, fuel bundles and fuel rods through dimensional measurements and visual exams. Measurements were taken on fuel components to verify models used by fuel vendors in various analyses and evaluations. These components are new designs and the measurements provide a verification that calculational models apply to the new designs.

Various General Electric equipment used for fuel inspections was installed along the east side of the Spent Fuel Pool (SFP) adjacent to the Fuel Preparation Machines (FPM). This NSE addressed the safety issues in connection with these tasks.

Two pieces of equipment required the use of the reactor building crane for installation in the SFP.

1. Channel Dimensional Measurement (CHAD) system for dimensional measurement of channels.

2. Combined Instrumentation Measurement System (COINS) for fuel rod oxide thickness and profilometry measurements.

The safety evaluation concluded that the end of cycle 10 fuel inspection did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A

JAF-SE-92-033, Rev. 1 EVALUATION OF RHR TO RADWASTE THROTTLE VALVE AND ISOLATION VALVE AS CONTAINMENT ISOLATION VALVES

This safety evaluation was written to document changes to the list of containment isolation valves in the Final Safety Analysis Report section 7.3 and Administrative Procedure 1.16. Specifically, the "A" side Residual Heat Removal to Radwaste drain isolation and throttle valves (10MOV-67 and 10MOV-57) will no longer be listed as containment isolation valves in either the FSAR section 7.3 or Administrative Procedure (AP) 1.16.

10MOV-57 and 10MOV-67 receive primary containment isolation signals but do not perform primary containment isolation functions. The design basis of 10MOV-57 and 10MOV-67 receiving primary containment isolation signals is to help prevent loss of water inventory from the RHR suction piping upon RHR shutdown cooling isolation.

The applicable portions of the JAF FSAR and Technical Specifications have been reviewed and no unreviewed safety questions result from these administrative changes. 10MOV-57 and 10MOV-67 are listed as exceptions to type C containment isolation valve testing in table 4.7-2 of the Technical Specifications, however these valves will be removed from this list by a separate safety evaluation.

The safety evaluation concluded that the administrative changes did not invclve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-069

JAF-SE-92-034, Rev. 0 UPGRADE OF CST LEVEL INSTRUMENT POWER SUPPLY

The purpose of this modification was to address NRC Information Notice, IN 84-09 which states that the level indication is recommended to achieve and maintain safe shutdown. The present Condensate Storage Tank (CST) level instrumentation loop was powered from 71RRACB8. In the event of a loss of offsite power during an Appendix R fire CST level indication would have been lost. The CST level instrumentation loop (33LT-101/LI-101A,B) is powered from the UPS bus, ensuring continuous CST level indication.
This modification also installed CST low level alarm indication to enhance the plant's conformance to the required provisions for direct readings of the process variable as delineated in Section III.L.2 of Appendix R.

The CST Level Instrumentation Loop components and the EPIC low level alarm do not have safety functions. The components will not degrade the Security Plan, Quality Assurance Program, Fire Protection System nor affect the Environmental Program of the plant.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A

JAF-SE-92-035 CO_2 PELLET CLEANING SYSTEM INSTALLATION AND OPERATION

The purpose of the installation of the CO_2 pellet cleaning unit was to provide improved capability for decontaminating tools, hardware, and other equipment.

Installation of the CO_2 pellet cleaning unit involves placing the unit on grade or cribbing and connecting the prepackaged unit to a 480 volt, 3 phase power supply.

No other services were required.

The CO_2 pellet cleaning unit consisted of a 14 ton liquid CO_2 storage tank, a compressed air delivery system, a CO_2 pelletizer, a delivery gun, a cleaning enclosure, and a support equipment enclosure. The unit was self-contained and transportable.

The system consisted of three main structures. There were two 40 foot long by 8 foot wide vans and one 40 foot flatbed trailer. The flatbed trailer contained a 14 ton CO_2 storage tank and the compressed air delivery system. One van contained the pellet cleaning enclosures, ventilation equipment, and contaminated storage. The other van contained the pelletizer, air drying equipment, electrical distribution equipment, and clean storage.

Installation and use of the CO_2 pellet cleaning unit did not affect the environmental impact of the plant nor did it involve an unreviewed environmental question. This was essentially because the unit did not contain enough radioactive material to cause a radiological environmental impact and contained no other material that could cause an environmental impact.

The safety evaluation concluded that the installation and operation of the CO_2 pellet cleaning units did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-88-186

JAF-SE-92-038, Rev. 0 GROUND FAULT DETECTOR REPLACEMENT STATION BATTERY CHARGERS, 71BC-1A,B

The purpose of this modification was to replace the original ground fault detectors in station battery chargers with new state-of-the-art ground detectors. This modification resolved a recommendation in OER 880467, Operating with Multiple Grounds in Direct Current Distribution System.

This modification replaced the original low sensitivity ground detectors in the two station battery chargers with new state-of-the-art ground detectors. This change did not change the basic functioning and operation of the battery charger or the DC distribution system.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A Temp. Mod.

JAF-SE-92-040, Rev. 0 RBCLC SIDESTREAM FILTER TO REMOVE MAGNETITE ACCUMULATION

The Reactor Building Closed Loop Cooling (RBCLC) System air operated containment isolation valves (15AOV-1[illegible] B 131A,B, 132A,B, 133A,B and 134A) had repeated problems due to binding during valve stroking. Investigation into this situation determined the cause of the valve problems to be a buildup of corrosion products on the valve internal surfaces. This corrosion product buildup is the result of the carbon steel piping materials in the RBCLC in combination with low oxygen chemistry which forms magnetite (black rust). The magnetite adhered to the valve stainless steel internals causing binding between the valve cage and the tefzel plug seal.

The solution to this problem was to remove the magnetite from the RBCLC system. To do this, a sidestream filter was installed by a temporary modification. This filter removed the magnetite from the system and prevented the valve binding problem from reoccurring.

The sidestream filter was installed to improve RBCLC system chemistry. By removing magnetite from the cooling water, the effects of corrosion product buildup on the RBC containment isolation valves will be greatly reduced. This will decrease the amount of maintenance on the valves and radiation dose received by mechanics.

The safety evaluation concluded that the temporary modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-91-307

JAF-SE-92-042, Rev. 0 FIRE RELATED PENETRATION SEAL REPLACEMENT FOR PENETRATION #S-4013

The purpose of this modification is to replace the existing non-fire rated seal at penetration S-4013 with an approved 3 hour rated fire seal.

This modification will replace the currently installed, non-fire rated seal at penetration S-4013 with a 3-hour fire rated seal consisting of products manufactured by 3M Fire Protection Products and Dow Corning. The penetration is located in the barrier between the East and West Cable Tunnels, and houses a 2 inch plastic (ABS) pipe which conveys drainage to the drain sump located in the East Cable Tunnel at the southwest end. The new seal is a functionally superior replacement of existing equipment, meeting or exceeding the design requirements imposed for the existing seal.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A

JAF-SE-92-043, Rev. EVALUATION OF RHR PUMP MINIMUM FLOW VALVES AS NORMALLY OPEN VALVE

This safety evaluation was written to document changes to the list of containment isolation valves in the Final Safety Analysis Report (FSAR) section 7.3 and Administrative Procedure (AP) 1.16. Specifically, the Residual Heat Removal pump minimum flow valves (10MOV-16A/B) will be listed as normally open vice normally shut containment isolation valves in both FSAR section 7.3 and AP 1.16.

The RHR minimum flow bypass piping does not communicate with the reactor vessel, but penetrates the primary containment as it returns from the common pump discharge to below the surface of the suppression pool. As a result, this process line is required by Technical Specifications to have a designated containment isolation valve. Therefore, 10MOV-16A/B not only provides the minimum flow requirements of the RHR pump, but is also listed as a containment isolation valve in AP 1.16 and FSAR section 7.3. These two tables also note that 10MOV-16A/B are normally shut. This safety evaluation provides justification to maintain 10MOV-16A/B in the normally open position.

The Residual Heat Removal (RHR) pump minimum flow valves (10MOV-16A/B) will be maintained in the normally open position vice closed position while the RHR system is in the LPCI standby mode. Administrative Procedure 1.16 and FSAR section 7.3 were revised to reflect this procedure change.

The safety evaluation concluded that the change from normally closed to normally open for valves 10MOV-16A&B did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A **HEAVY LOAD EVALUATION**

JAF-SE-92-047, Rev. 0 **LOAD PATH AND FLOOR LOAD EVALUATION FOR DRYWELL HEAD**

This evaluation provides a review of safety issues associated with temporarily locating the drywell dome north of the SE equipment hatch on Refueling floor. Scope of this nuclear safety evaluation was limited to evaluation of permanent structural members and components for integrity under the static load of the drywell dome.

Maintenance required an alternate temporary location other than the designated temporary location of the drywell dome. The drywell dome was stored at location H which is the permanent storage location. The Reactor Building Crane, Main Hook with rated capacity 125 tons was used. The drywell dome (weight 43 tons) was lowered on three existing connected pedestals consisting of 12 inches diameter pipe and 14 inches square plate each. Three A-36 steel bearing plates, 24" x 24" x 1" was centered directly below each pedestal.

The drywell dome was located in the desired location with additional base plates located under the pedestals so that loads were transferred directly on existing beams B503 and B505 and B506.

The safety evaluation concluded that the new temporary location for the drywell dome did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A

JAF-SE-92-048, Rev. 0 CORE SPRAY SYSTEM INSTRUMENTATION INSTALLATION AND VIBRATION TEST

The purpose of this safety evaluation covered Phase I of the Core Spray (CS) Piping System vibration measurements and analysis to determine the piping system's design adequacy for the combination of test and design loads. The vibration measurements and analysis were performed for Loop A in West Crescent Area. The results shall be applicable for both Loops A & B in West and East Crescent areas.

The vibration test performed on the discharge piping of core spray pump 14P-1A recorded dynamic data to be used in evaluating the piping stress levels and vibration characteristics during core spray pump testing and actuation. This data was recorded using accelerometers, strain gauges and pressure transducers attached to the pump discharge piping.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-120

JAF-SE-92-049, Rev. 0 TEST SWITCHES FOR SDIV VENT AND DRAIN VALVES

The purpose of the test switches is to exercise and measure the stroke times of the SDIV vent and drain valves in accordance with ST-20L by way of their isolation AC solenoid valves. The previous method of testing SDIV vent and drain valves consisted of lifting leads to de-energize the isolation AC solenoid valves, 03SOV-31A&B, thereby venting the air from the SDIV vent and drain valve air header and closing the valves. Lifting energized leads inside the 09-15 and 09-17 panels, RPS-A&B, near numerous other primary isolation circuits was not desirable with respect to plant availability and personnel hazard. The test switches ensure full ASME Section XI code compliance for the IST program so no relief is required from the NRC.

The test switches control the SDIV vent and drain isolation valves 03SOV-31A&B solenoid power circuit. When both of the valve solenoids are energized the SDIV vent and drain valve air header is pressurized by the scram air system which holds the vent and drain valves open. When both of the valve solenoids are de-energized the SDIV vent and drain valve air header is vented which closes the vent and drain valves. The test switches are key lock devices for administrative control.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A TEST: STP-11

JAF-SE-92-050, Rev. 0 SPECIAL TEST TO DETERMINE EFFECTS OF ACCUMULATOR REMOVAL ON SLC SYSTEM VIBRATION

The purpose of this test is to determine the effects of the Standby Liquid Control (SLC) pump discharge accumulators on system vibration.

The SLC system was operated in a recirculation line up (test tank to test tank) with the SLC pump discharge accumulators installed and removed.

The SLC pumps were required to pump against maximum system normal operating pressure (1275 psig) by throttling at the SLC pump discharge header to recirculation tank isolation valve (11SLC-26). Vibration data was collected at various locations on the SLC system piping by SWEC.

If the accumulators have negligible contribution to the suppression of system vibration, the accumulators will either be removed or the internal gas bag and valve assembly will be maintained in a deflated condition.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-91-305

JAF-SE-92-052, Rev. 1   LPCI ALTERNATE POWER SUPPLY CIRCUIT MODIFICATION

The purpose of this modification was to provide a control scheme which would enable the plant operators, from the control room, to isolate the LPCI injection valve bus independent power supplies and connect a maintenance bypass (to be renamed Alternate Feed) from another safety related emergency MCC in the same safety division to the valve bus. This fulfilled the objective of the operators having full control over the power sources for the LPCI injection valve bus even after access to the reactor building had been restricted due to post-accident postulated radiation dose levels.

The existing three position, key-operated 1A-1IPSA(B)02 switches (AC Input to LPCI Mov Indep Pwr Sups) at 09-08 were replaced with new four position switches to provide the additional capability of opening LPCI Inverter output contactors 71MCC-153-G2B (MC) and 71MCC-163-C2B (MC) (currently spared in 71MCC-153-G2B and 71MCC-163-C2B) and closing existing Maintenance (Alternate) Feed Contactors 71MCC-153-G2A (MC) and 71MCC-163-C2A (MC) from the control room.

The modification rerouted the inverter output cables to the spare size 4 contactors, located in 71MCC-153-G2B and 71MCC-163-C2B. The contactors output were connected by internal MCC wiring to 71MCC-155-OH3 and 71MCC-165-OB3.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-128

JAF-SE-92-053, Rev. 0 REPLACEMENT OF CONTROL ROD BLADES AT EOC10

Replacement of eight original equipment CRB's which had reached the end of their service life. The CRB's were identified by core position. Service life criteria requires CRB replacement to maintain adequate control strength.

The replacement CRB's were of two types; there were four General Electric CRB's called MARATHON and four ABB CRB's called CR-82M. Both types incorporate various improvements in CRB design over original equipment, including use of non-cobalt bearing pins and rollers, hafnium absorber to extend CRB life, and high-purity stainless steel to contain the absorbing material.

A change to FSAR Section 3.4 was required.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-148

JAF-SE-92-054, Rev. 0 INSTALL BLANK INSERT AT RHR TO FPC SUCTION CROSSTIE FLANGE

This safety evaluation assessed the effects on plant safety of installing a Residua Heat Removal System (RHR) Assist to Fuel Pool Cooling and Cleanup (FPC) suction crosstie isolation blank on line 10-8"-W20-152-039.

The RHR Assist mode of the RHR system uses the RHR to FPC system crosstie piping to provide cooling water to the Spent Fuel Pool (SFP) when a full reactor core is off-loaded to the SFP or when FPC components are out of service for maintenance.

The original design of the RHR to FPC crosstie included a spectacle flange which was removed by modification M1-89-103 based on the apparent redundancy of an installed isolation valve in the FPC system. This modification installed a blank in this RHR to FPC suction crosstie flange. The blank was designed to meet the original design specification. Installation of the blank did not adversely affect the seismic qualification of the piping system because the relative decrease in weight from the spectacle flange to the blank is not significant.

This modification restores the RHR to FPC suction crosstie piping to an acceptable and previously evaluated configuration. The installation of the new blank meets the system requirements as described by the original system design specifications.

A change to FSAR Section 9.4 figure 9.4.1 is required.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A **TEST: POT 10-T**

JAF-SE-92-055, Rev. **JUSTIFICATION FOR PERFORMANCE OF PREOP TEST 10-T**

The purpose of this pre-operational test is to obtain data in order to verify the functional capability of the RHR pump discharge flow restriction orifices (10RO-107A through D) to achieve minimum Low Pressure Coolant Injection flow requirements following completion of modification M1-91-242.

Modification M1-91-242 reduced the size of the RHR pump runout orifices to limit maximum Low Pressure Coolant Injection (LPCI) and Shutdown Cooling mode flow rates. After installation of the new orifices, this pre-operational test obtained system process data to verify successful design of the orifice and to aid in the development of subsequent surveillance tests.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A

JAF-SE-92-057, Rev. 0 TEMPORARY STORAGE OF RADIOACTIVELY CONTAMINATED MATERIALS ON SITE

This nuclear safety evaluation was performed to document whether the temporary storage of radioactively contaminated materials on site constituted an unreviewed safety question.

The materials in question are various types of liquid machine oils and filter media (activated charcoal, desiccant and epoxy paint). The liquids may or may not contain small amounts of water. All material is contained in 55 gallon shipping containers thus not readily transportable to the environment.

The materials are stored in trailers located on the spare transformer foundation on the West side of the plant (North of the 115 KV switchyard). The oil is potentially combustible, flammable/explosive material. Nuclear safety evaluation (JAF-SE-92-009, Rev. 1) was completed which addressed the environmental effect of a potential spill or container failure.

The safety evaluation concluded that the storage of the contaminated oil on the Northwest Side of the site did not constitute changes to technical specifications incorporated in the license or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-111

JAF-SE-92-060, Rev. 0 ISOLATION OF RHR VALVES FOR RC BUILDING FIRE

The modification provided selector switches on the 09-3 panel in the control room (CR) to allow the CR operator to bypass logics and control the operations of the Residual Heat Removal (RHR) valves 10MOV-16A, 10MOV-16B, 10MOV-25A, 10MOV-25B, 10MOV-27A, 10MOV-27B, 10MOV-66A and 10MOV-66B during an Appendix R fire in the Reactor Building. A fire could cause the valves to operate spuriously.

In addition this modification re-routed portions of cable 1RHRBBC120 to comply with the separation requirements of Appendix R.

The selector switch is key-locked, a two position (normal and bypass) device with the key removable in the normal position. Except for Inboard Injection Valves (10MOV-25A, 10MOV-25B), the switch has three (3) normally open (n.o.) and 3 normally close (n.c.) contacts. The inboard injection valve switch has 4 n.o. and 4 n.c. contacts.

A white indicating light was provided for each selector switch to indicate when the switch is in the bypass/override position.

FSAR Sections 7.3 and 7.4 and the applicable logic diagrams (Fig. 7.4.8) will be revised to reflect the addition of the bypass switches. The FSAR change will not affect the existing safety evaluation.

The safety evaluation concluded that the modification did not involve a change in the Technical Specifications or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-92-068

JAF-SE-92-061, Rev. 0 EDG JACKET COOLER RETURN LINE REROUTING

The purpose of this modification was to reroute the Emergency Diesel Generator (EDG) jacket cooling water common return lines 46-8"-WES-151-10 and 46-12"-WES-151-9 such that its flow does not mix with the water in the Emergency Service Water (ESW) pump suction. The EDG jacket cooling water return flow from both trains was routed to the ESW pump bay "A". This could have potentially increased the ESW supply water temperature above maximum design lake temperature during Emergency Diesel Generator operation.

Therefore, the EDG jacket cooling water return flow was rerouted to the Circulating Water (C.W.) Discharge Tunnel via two independent six inch lines, one 6" line for each EDG train. This will prevent the return flow from directly mixing with the water in the ESW pump bay. This modification also removed check valves 46ESW-6A and 6B, which were no longer needed. In addition, the flow bypass lines 46-8"-WES-151-116A and 116B, which are connected to the EDG return line, were separated into two independent lines discharging into their respective pump bays providing operational flexibility. Existing piping and supports were demolished as required.

Fig. 9.7-2 and Sec. 9.7.1 Emergency Service Water System of the UFSAR will be revised to incorporate changes due to modification. UFSAR Section 12.-5 - Class II Structures will also be revised to indicate that the gates and trash racks are analyzed to withstand seismic loads.

The safety evaluation concluded that the modification did not involve a change in the Technical Specifications or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-121

JAF-SE-92-063, Rev. 3 RELOCATION OF FIRE DOOR 76FDR-E-272-10

Existing fire door 76FDR-E-272-10 did not achieve a proper fire barrier between the Turbine Building and Screenwell House. The fire door was located on the Screenwell side of the opening, which has steel grating floor structure. Hence, this fire door did not provide an adequate fire barrier as installed.

This modification replaced existing fire door 76FDR-E-272-10, which was located on the Screenwell side of the opening between the Turbine Building and Screenwell House, with a new 3-hour fire door. The new fire door was located on the Turbine side of the opening, thus providing an adequate fire barrier.

This modification restored the integrity of the fire barrier boundary to comply with the requirements of 10CFR50 Appendix R. A Fire Protection/Appendix R Compliance review has been conducted. The results of the review show that this modification did not represent an unreviewed safety question, impact Safe Shutdown Systems, equipment or access and concluded that a detailed Fire Protection Program Analysis was not required.

The safety evaluation concluded that the modification did not involve a change in the Technical Specifications or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-122

JAF-SE-92-064, Rev. 0 FIRE PROTECTION INSIDE LOOP HEADER ISOLATION VALVE OPERATOR REPLACEMENT

The manual operator for valve 76FPS-283 penetrated a ceiling (barrier) between fire areas. The fire barrier penetration number S3019 did not provide a fire rating equivalent to or higher than the required rating of the barrier. By replacing the valve operator with a 90° angle operator the existing penetration could be sealed and the integrity of the fire barrier boundary would be restored to comply with the requirements of 10CFR50 Appendix R. A Fire Protection/Appendix R review was performed in accordance with Engineering Standard IES 4.2 and concluded that a detailed Fire Protection Program Analysis was not required.

This modification replaced the existing operator on valve 76FPS-283 with a new 90° angle operator to allow sealing of existing fire barrier penetration S3019. The sealing of this penetration restored the integrity of the fire barrier boundary to comply with the requirements of 10CFR50 Appendix R.

The support for the new 90° angle operator for valve 76FPS-283 was seismically designed in accordance with requirements for JAF and qualified per Calculation JAF-CALC-FPS-00491.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-171

JAF-SE-92-066, Rev. 2 3 HOUR FIRE SEALS FOR CLOSED BONDSTRAND PIPE SYSTEMS

The purpose of this modification is to replace or modify the existing non-qualified seals at the penetrations in question with qualified, 3 hour rated penetration seals.

The seals installed in the penetrations in question consisted of a minimum of 10" of RTV silicone foam, installed in the annular space around various sizes of bondstrand fiberglass pipe. This design was not specifically qualified by test for bondstrand pipe (seal was qualified for steel pipe). These seals were upgraded to a qualified 3 hour design by increasing the foam or elastomer thickness to a minimum of 12", and providing a protective cover over the penetration on both sides of the wall (on the bottom for a ceiling). The protective cover is 3M's CS-195 intumescent sheet, doubling as the foam dam. Exceptions were for penetrations WS-60, which has an elastomer depth greater than 12" and a protective cover on only one side of the penetration and S-2135, having 22" of elastomer and no CS-195. Both exceptions were due to ALARA concerns. These penetration seals are detailed on NYPA drawing 11825-FP-119A & B.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-176

JAF-SE-92-067, Rev. 0 LONGER COVER STUDS FOR TORUS ACCESS PENETRATION "B"

Closure studs in the "B" Torus manway cover were too short to permit complete thread engagement in both nuts located at either end. The ends of several studs were up to 1-2 threads below the nut surface. This modification installed longer studs to facilitate full thread engagement with all closure nuts.

The replacement of 6½ inch long manway closure studs with 7¼ inch studs was an acceptable method to achieve complete thread engagement with the associated nuts.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A

JAF-SE-92-068, Rev. 0 NUCLEAR OPERATIONS DIVISION RE-ORGANIZATION

The purpose of this safety evaluation was to describe and evaluate a management reorganization of the Nuclear Operations Division, and to document that the organizational changes did not involve an unreviewed safety question.

The reorganization involved the reassignment of duties and position responsibilities by function. These changes did not in any way alter the Power Authority's commitment to maintain a management structure that contributes to the safe operation and maintenance of the plant.

The reorganization of the Nuclear Operations Division by function was accomplished in the following manner. The responsibilities of the Directors - Nuclear Operations and Maintenance were divided along the functional lines of Operations and Maintenance, rather than the previous territorial responsibilities of PWR and BWR activities. The two Directors now have the responsibility for operations and maintenance activities at each of the Authority nuclear facilities respectively. The duties and responsibilities of the Division were not changed. They were reallocated along functional lines to the appropriate Director within the Division.

The reorganization of the Nuclear Operations Division is designed to restructure the division along functional lines. The changes include position title changes and the reassignment of position responsibilities. 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 plants, and do not involve an unreviewed safety question.

The safety evaluation concluded that the Nuclear Operations Division re-organization did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-181

JAF-SE-92-069, Rev. 0 CORE BORING AND HILTI BOLT DRILLING FOR MOD F1-92-068

The purpose of this modification was to core bore (12) 10 inch diameter holes and drill anchor bolt holes for 32 supports for the Emergency diesel Generator (EDG) Jacket Cooler Discharge Return Line Rerouting Modification F1-92-068 prior to disabling the operation of the EDG before the scheduled tie-in.

This modification completed processes for pre-installation of the portions of core boring, supports and piping needed to support the tie-in schedule of Modification F1-92-068 and core bored (12) 2 inch diameter temporary holes through the ESW Pump Room A floor for scaffolding and which were re-sealed when scaffolding was removed.

This modification core bored and drilled anchor bolt holes adjacent to the existing piping without disturbing the systems operation for the installation of the new, rerouted emergency diesel generator jacket cooler discharge piping and supports for lines 46-6"-WES-151-116A, 46-6"-WES-151-116B, 46-6"-WES-151-10, and 46-6"-WES-151-132.

The implementation of this modification required the three (3) screenwell gates to be located to the center gate slots. Also the removal, temporary storing and supporting of the wall mounted storage cabinet 76 CAB-1, maintaining its intended function and final installation north on the east cable tunnel wall inside ESW Pump Room "A".

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-90-340

JAF-SE-92-070, Rev. 0 REPLACE AIR SUPPLY WITH NITROGEN TO VACUUM BREAKER ISOLATION VALVES

This mod required tapping into the existing N_2 to Containment Instrumentation Line 1"-N-151 (EL 285'-6") downstream of 27FE102 with a 1" tee and 1/2" isolation valve. The pipe root connection was immediately reduced to 1/2" O.D. Tubing (Stainless Steel ICN8) and routed approximately 7 feet to 27AOV-101A (SOV101A) reducing to 3/8" O.D. tubing and routing an additional 8 feet to 27AOV-101B (SOV 101B). The tubing was routed and supported in accordance with plant seismic guidelines.

The design change fulfilled the objective of the Work Request #087663 and improved the reliability of existing safety-related components (AOV-101A/B) by replacing a non-safety related pneumatic supply with a safety-related one. The source of the safety-related pneumatic supply is the nitrogen supply to containment instrumentation. The Containment Instrumentation nitrogen can be supplied by either one of the two redundant containment Atmospheric Dilutions (CAD) trains as discussed in the FSAR (Section 5.2.3.8).

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-92-091

JAF-SE-92-071, Rev. 2 DEACTIVATION OF THE RHR HEAD SPRAY SYSTEM

The purpose of this modification was to deactivate the reactor vessel head spray portion of the Residual Heat Removal (RHR) System. The reactor vessel head spray system was designed to spray RHR System water through a nozzle in the reactor vessel head to promote rapid cool down of the vessel head region during shutdown. It was anticipated in the design phase of the plant that vessel cooldown and head removal would be critical path activities, and that rapid head cooldown would reduce outage time. However, as vessel cooldown and head removal are not critical path at JAF, the head spray system (which is an optional system) was not being utilized. Further, the head spray system was not required for shutdown cooling, and design documentation for the RHR system indicated that the head spray system did not perform any safety-related function.

The objective of this modification was to eliminate system maintenance and surveillance testing of the valves in this unutilized piping sub-system. This modification also eliminated the maintenance, ISI and surveillance testing, and attendant man-rem exposure associated with the retired or removed equipment.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-150

JAF-SE-92-072, Rev. 0 CO_2/FIRE DAMPER REPLACEMENT FOR ELECTRIC BAY EXHAUST CARBON DIOXIDE EXHAUST DAMPERS

This modification replaced two existing, non-rated, CO_2 actuated fire dampers in the Turbine Building Ventilation System (System NO. 67). Damper 67CD-4 is located in the south wall of the East Electrical Bay Switchgear Room (Fire Area 2), and damper 67CD-3 is located in the south wall of the West Electrical Bay Switchgear Room (Fire Area 1C). These curtain-type CO_2/fire dampers are located in 3-hour fire barriers, and serve to isolate the Electrical Bay Switchgear Rooms upon CO_2 system actuation to maintain CO_2 concentration, as well as maintain the integrity of the 3-hour fire barriers. Occurrence Report No. 92-114 documents NRC findings which concluded that these dampers were acceptable for the above described functions and were also considered inoperable. Neither damper had a UL fire rating equivalent to the rating of the Appendix R wall in which it was installed, nor were they provided with the proper thermal expansion capabilities, as required by NFPA and SMACNA Codes. In addition, damper 67CD-3 was found to be damaged. This modification replaced these dampers with new approved/rated dampers and restored the dampers to an operable status.

In order to accommodate the existing security grate and louvered grill, a curtain type damper was used for both replacement dampers. The new dampers are Ruskin Model NIBD23 UL approved, three-hour fire rated, curtain type fire dampers.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-92-151

JAF-SE-92-073, Rev. 1 RELOCATION OF EDGSWGR ROOM NORTH FIRE PROTECTION CO_2 PANEL

This modification relocated fire protection panel 76CO_2-PNL-8, associated pushbutton 76PB-FPS-8 and subpanel 76EMPC-8 from the Screenwell Area (Zone SH-13/Fire Area 1B, El. 272'-o", Col. 24, Line A) to the Turbine Building (Zone RB-2/Fire Area 1E, El. 272'-0", Col. 23, Line A). Junction box JB-FPS22 in the Turbine Building Cable Tunnel-East (El. 260') was replaced with a Terminal Box.

Three new wall penetrations were added, two between the Turbine Building and Screenwell Area and one between the Turbine Building and Cable Tunnel-East.

Standard hardware was utilized in remounting panels, tubing, and conduit in accordance with standard practice. Fire Barrier penetrations were opened and sealed in accordance with standard practice.

The modification provided corrective actions in response to an Appendix R Evaluation.

Relocation of panel 76CO_2-PNL-8 and associated pushbutton and subpanel to the Turbine Building fulfilled the objective of compliance with 10CFR50 Appendix R for this equipment.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-92-166

JAF-SE-92-074 HPCI VALVE CIRCUITS MODIFICATION

The purpose of this modification was to modify the circuits and/or reroute cables associated with HPCI isolation valves 23MOV-15 and 23MOV-16 and HPCI turbine stop valve 23HOV-1. This was necessary to (1) isolate the HPCI turbine steam supply line to shutdown the HPCI turbine and (2) control the operation of the HPCI isolation valves 23MOV-15 and 23MOV-16 to isolate the HPCI steam supply line or maintain HPCI turbine operation, during plant fires. This modification was implemented to comply with the requirements of 10CFR50, Appendix R. The QA class for this modification is QA Cat. I.

This modification upgraded the HPCI systems ability to (1) isolate the HPCI turbine steam supply line to shutdown the turbine and (2) control the operation of HPCI isolation valves 23MOV-15 and 23MOV-16 to isolate the HPCI steam supply line or maintain HPCI turbine operation during plant fires.

The modification met all design and material requirements for a QA Category I installation.

The following portions of the FSAR required update, Section 6.4.1, 7.4.3.2.5 and Fig. 7.4.3.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION:

JAF-SE-92-075 NUCLEAR ENGINEERING DIVISION REORGANIZATION

The purpose of this safety evaluation was to describe and evaluate a management reorganization of the Nuclear Engineering Division, and to document that the organizational changes did not involve an unreviewed safety question.

The reorganization involved reassignment of position responsibilities to focus accountability. In addition, one Section had been recognized to assume new responsibilities and two new groups have been added.

The reorganization of the Nuclear Engineering Division was designed to better concentrate accountability for design changes in support of the plant needs, to enhance Division capability with the establishment of the Site Engineering and Planning and Scheduling Groups, and to allow a concentration on configuration management issues. The changes did not alter the Power Authority's commitment to maintain a management structure that contributes to the safe operation and maintenance of the plant, and did not involve an unreviewed safety question as defined in 10CFR50.59.

Subchapter 13.2 of the JAF FSAR requires revision to reflect the reorganization.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-188

JAF-SE-92-077 STATOR WINDING COOLING WATER SYSTEM EXPANSION TANK VENT MODIFICATION

The present stator winding cooling water expansion tank vent arrangement was believed to be contributing to a hydrogen vapor seal which could prevent an air exchange in the vapor space of the tank and cause the cooling system to operate "Air Depleted". This modification to the vent configuration assures hydrogen venting and cooling water air saturation by providing an additional vent path.

This change modified existing vent pipe arrangement by adding a tie into the existing upper horizontal portion of the tank vent line and route the line to a tie in point with the 1 1/2" vertical vent line above the existing tie in point and provided an isolation valve test connection in the new vent path line.

This modification addresses a deficiency with the water chemistry in the Stator Winding Cooler Water (SWC) System. This system is a Q.A. Category II/III system that provides deionized cooling water to the main generator stator winding and exciter static rectifiers.

No possible safety hazards were associated with the modification and its installation since no safety related or system important to plant safety were affected. Installation was performed while the plant was in a cold condition. Associated equipment at the proximity of the installation had been analyzed for possible failure due to installation error or inadvertent damage. No safety hazard was presented.

No portion of the FSAR or Technical Specification was affected by the installation described. All potential accidents had been considered.

This modification was initiated to improve chemistry reliability of the SWC System. The installation of the new vent in accordance with the proposed design and plant procedures did not pose a safety concern.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-91-204

JAF-SE-92-078 REPLACE FIBERGLASS DISCHARGE HEADER OF THE PLANT WATER TREATMENT SYSTEM TRANSFER PUMPSWITH STAINLESS STEEL

The purpose of this modification was to satisfy the plant requirement for a more reliable discharge header off of the plant Water Treatment System transfer pumps (42P-4A&B). The existing fiberglass header had been in service for a number of years and had developed cracked fittings and piping joints. The fiberglass header was replaced with stainless steel. The transfer pump discharge gate and check valves were replaced due to their condition and lack of spare parts.

Valves were added to the Turbine Building Closed Loop Cooling (TBCLC) System and Reactor Building Closed Loop Cooling (RBCLC) System to allow connections for providing make-up to those systems when the transfer pumps are out of service or a source of demineralized water is required.

The following portion of the FSAR required update: Fig. 9.10-4.

This modification to the Water Treatment System was consistent with design requirements as detailed in the JAF FSAR Sections 9.10 Make-up Water Treatment System and 16.5 Pressure Integrity of Piping and Equipment Pressure Parts.

JAF Technical Specifications were not affected by this modification because the transfer pumps and associated discharge header and valves have no safety related function.

In conclusion, this modification to the plant Water Treatment System (Demineralized Water Storage and Transfer System) did not conflict with the objectives and design bases as stated in the JAF FSAR nor result in changes to the Technical Specifications. Safety-related structures, systems, or components were not adversely affected by this modification.

This modification did not constitute an unreviewed safety question pursuant to 10CFR50.59.

MODIFICATION: M1-91-123

JAF-SE-92-079 HPCI SYSTEM ENHANCEMENTS

The following modifications were performed to enhance the operability of the HPCI system and to reduce the amount of exposure required during the performance of surveillance testing.

- Replace HPCI Lube Oil Pressure Gauges
- Add HPCI Booster Pump Discharge Pressure Gauge 23PI-319
- Replace Vacuum Gauges on HPCI Turbine (23PI-316, 37A,B, 318A,B)
- Installation of Break Flanges HPCI Booster Pump Suction Safety Valve (23SV-34)
- Modify Coupling Guards
- Modify HPCI & RCIC Testable Check Valves
- HPCI Temperature Elements (23TE-114A&B) Install shield around element tips
- The following portions of the FSAR required update: Fig. 4.7-1, 7.7-2, 7.4-2, 7.4-3 and 9.11-1

The operation of the HPCI/RCIC system remained unchanged. The modifications were for the purpose of facilitating operator and maintenance staff in routine activities in and around the HPCI Turbine and Pumps. These changes enhanced the system by providing additional pressure indication at 23P-1B discharge, providing larger gauges and centrally locating gauges at the HPCI turbine and providing a means of testing 23SV-34. The changes to and addition of coupling guards at the HPCI pumps improved personnel safety. The modification to the HPCI/RCIC testable check valves provided positive indication of the valves functional capability. The addition of the shields to the HPCI temperature elements will protect the temperature elements from physical damage. The design and installation of these modification are consistent with practices accepted in the FSAR.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-134

JAF-SE-92-080 ISOLATE CABLE 1ESWBBC098 FOR A CONTROL ROOM FIRE

This modification improved the control circuitry of ESW pump, 46P-2B, and made the operation of the pump more reliable from the Auxiliary Shutdown Panel 25ASP-3, in the event of a Control Room evacuation.

In the event of Control Room evacuation, the Emergency Service Water (ESW) pump 46P-2B should remain operable from the Auxiliary Shutdown Panel 25ASP-3 located outside the Control Room. Appendix R Control Room Fire Analysis indicated that this pump could become inoperable from the Auxiliary Shutdown Panel 25ASP-3 in the event of a fire in Fire Area 07.

This modification did not change any operating features or operating sequence or procedure and, hence, it did not involve a change to the plant Technical Specifications or FSAR.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-92-131

JAF-SE-92-082, Rev. 1 REROUTE/WRAP CABLES FOR APPENDIX R FIRES

The purpose of modification F1-92-131 was to reroute four (4) cables affected by Appendix R postulated fires from various fire areas and wrap one (1) conduit, pull boxes and supports with an approved fire protective material.

No electrical components (except for cable, conduits, junction boxes and splices) were added or deleted as a result of this QA Category I modification.

This safety evaluation confirmed that rerouting and splicing of existing (and new) cables did not affect the functions of the following equipment:

1. 125V DC MCC 71BMCC-2
2. CO2 Fire Suppression System Panel 76CO2-PNL-7
3. RHR Pump 10P-3D.

This safety evaluation reviewed the potential impact on plant safety due to the reroute of cables 1DMSBBK015, 1FPSNNC233, 1FPSNNC235 and 1RHRDBH004, and the fire wrap protection system installed on conduit 1CK201BD1, which contains cable 1ABVBBK055, for Appendix R fires. Based on the Review and Analyses, it can be concluded that this modification did not adversely impact the operation of 125V DC MCC 71BMCC-2 and its associated loads, CO_2 Fire Suppression System Panel 76CO2-PNL-7 and RHR Pump 10P-3D and all equipment operates as before.

The implementation of this modification did not conflict with the design basis of existing cable/conduit routes and circuits which they are a part of as stated in the FSAR, it did not result in changes to the JAFNPP Technical Specifications, and did not adversely impact any safety related and environmentally qualified structures, systems and components.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: TEST: POT-71AB

JAF-SE-92-084, Rev. 1 REROUTE/WRAP CABLES FOR APPENDIX R FIRES

The purpose of this safety evaluation was to examine the impact of this pre-operational test on plant safety. The test was conducted to verify that rerouting and splicing of cables has not affected the functions of the equipment:

1. The rerouting and splicing of cable 1DMSBK015 did not functionally affect the operation of 125V DC MCC 71BMCC-2 and its associated loads.

2. The installation of new cables 1FPSNNC898 and 1FPSNNC899, spliced to existing cables 1FPSNNC233 and 1FPSNNC235, respectively, did not affect the function of Emergency Switchgear Room CO_2 Fire Suppression System (76CO2-PNL7).

3. Rerouting and splicing of cable 1RHRDBH004 did not affect the operation of Residual Heat Removal Pump 10P-3D.

As no change was made to Control Circuits or operation of equipment, the FSAR or Technical Specification was not affected. This test procedure did not conflict with or degrade the design basis or function of any other plant system or components and hence it was performed without compromising the safety of the plant.

The safety evaluation determined that the test was not described in the FSAR and concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-195

JAF-SE-92-085 AUXILIARY BOILER VENT REMOVAL

The purpose of this modification was to remove the existing atmospheric vent from the auxiliary boiler system. The vent removal included the vent pipe, associated drains, isolation valves and vent muffler. One Auxiliary Boiler (87AB-1A) has been retired, and the other (87AB-1B) has been removed. These items, including the muffler which was located outside the east wall of the Auxiliary Boiler Room (ABR), were radiologically contaminated. The muffler and exterior piping was removed due to radiological requirements prior to construction of the new Administration Building, which will be located adjacent to the ABR east wall. The vent and drain piping inside the ABR was also removed to reduce exposure rates in the ABR.

This removal lowered the man-rem dosages outside the east wall and on the interior of the Auxiliary Boiler Room. This vent system was a QA Category II/III non-safety related system.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-92-078

JAF-SE-92-086 ISOLATION OF MSIV & SRV VALVES FOR APPENDIX "R" FIRES

The purpose of this Nuclear Safety Evaluation (NSE) was to examine the impact on plant safety of the modification to provide isolation of the four outboard Main Steam Isolation Valves and the eleven Safety Relief Valves thus satisfying Appendix "R" concerns.

The Safety Relief Valves (SRVs) consist of a combination of seven Automatic Depressurization System Valves and four Manual Depressurization System Valves. A Control Room fire could induce a hot short in these circuits, with the resulting spurious opening of these valves possibly leading to core uncovery.

A new Auxiliary Shutdown Panel (25ASP-4) was installed outside the control Room which allows the operators to isolate the control circuitry for the MSIVs in the event of a control Room fire. Valve position indication also is available.

Another Auxiliary Shutdown Panel (25ASP-5) was installed outside the Control Room which allows positive isolation of the SRVs thus negating the possibility of a spurious opening due to Control Room or Reactor Building fire.

Auxiliary Shutdown Panels (25ASP-4 and 25ASP-5) are vital equipment for the safe shutdown of the plant. These panels are locked and alarmed and monitored for entry by Security and Operations.

As the MSIVs and SRVs are both Safety Related systems, this modification is classified as QA Category I.

The JAFNPP FSAR, Sections 4.4, 4.6, 6.4, 7.3, 7.4 and 14.5, were reviewed and revisions to Section 14.5 have been submitted.

It is concluded, therefore, that this modification did not involve an unreviewed safety question. The implementation of this modification did not conflict with the design basis of the MSIVs as stated in the FSAR, nor result in changes to the Technical Specifications, nor adversely impact any safety-related or environmentally qualified structures, systems or components.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: TEST: POT-29C

JAF-SE-92-087 ISOLATION OF MSIV FOR APPENDIX "R" FIRES

The purpose of this Nuclear Safety Evaluation (NSE) was to examine the impact of Pre-Operational Test POT-29C on plant safety. The test was conducted after the Main Steam Isolation Valve portion of modification F1-92-078, was completed and before the refueling outage was over. The test verified that the four outboard Main Steam Isolation Valves (29AOV-86A, -86B, -86C and -86D) could be isolated from the Main Control Room at the Auxiliary Shutdown Panel 25ASP-4.

This pre-operational test verified that the four Outboard Main Steam Isolation Valves could be operated from the Control Room Panel 09-3 and could be isolated from the Control Switches on the Auxiliary Shutdown Panel 25ASP-4 and bring the MSIVs to a closed position in case of a Control Room Fire. The test was conducted during the refueling mode of the plant, when the reactor was depressurized in order to verify the functioning of the MSIVs. This test did not involve an unreviewed Safety Question and did not involve a change in the Technical Specification. This test procedure did not conflict with or degrade the design basis or function of any other plant system or components and was performed without compromising the safety of the plant.

The safety evaluation determined that the test was not described in the FSAR and concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-92-132

JAF-SE-92-088 ISOLATION OF DE-ICING HEATERS FROM CONTROL/RELAY/CABLE SPREAD ROOM FOR APPENDIX "R" CONCERNS

The purpose of the modification was to allow the operation of the Intake Structure De-icing Heaters without being impacted by a fire in the Control Room or the Relay Room. De-icing Heater Control Components located in the Control Room and Relay Room was removed, thus deleting the control of heaters from the Control Room and making the control available only at the local control panel at the screenwell area.

This modification allows the heater control circuit to be independent of a Control Room or Relay Room fire. Control Room indication circuitry was also removed to prevent fire induced failures during a Control Room fire from affecting operation of the de-icing heaters.

This safety evaluation reviewed the potential impact on plant safety due to the isolation of control of de-icing heaters from the Control/Relay/Cable Spread Rooms. The implementation of this modification did not conflict with the design bases for the de-icing heaters as stated in FSAR, nor result in changes to the Technical Specifications, nor adversely impact any safety related or environmentally qualified structures, systems or components.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: **TEST: POT-36E**

JAF-SE-92-089 **ISOLATION OF DE-ICING HEATERS FROM CONTROL/RELAY/CABLE SPREAD ROOM FOR APPENDIX "R" CONCERNS**

The purpose of this nuclear safety evaluation was to examine the impact of pre-operational test POT-36E on plant safety. The test was conducted after the installation of modification F1-92-132 was completed and before the refueling outage is over. The test verified that the intake structure heaters could be operated from the Local Control Panels (36TRH 6A & 6B) after the installation of the modification. This modification was classified QA Category I.

This Pre-Operational Test POT-36E for the Intake Structure De-icing Heaters was conducted during the refueling mode of the plant, in order to verify the operability of the Heaters and their associated circuits. This Test Procedure did not conflict with or degrade the design basis or function of any plant system or components and was performed without compromising the safety of the plant.

The safety evaluation determined that the test was not described in the FSAR and concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A TEST: POT-46E

JAF-SE-92-091, Rev. 0 ISOLATE CABLE 1ESW3BC098 FOR A CONTROL ROOM FIRE

The purpose of this nuclear safety evaluation is to examine the impact of Pre-operational Test (POT-46F) on plant safety. The test was conducted after the Installation of Modification M1-92-134 was implemented. The test verified that the Emergency Service Water Pump 46P-2B can be operated from Auxiliary Shutdown Panel 25ASP-3, as well as Control Room Panel 09-6. As this modification involved changes to the Control Circuit for the ESW Pump 46P-2B, only Control Circuitry was tested for proper functioning. This modification is QA Cat. I.

The implementation of modification M1-92-134 and the pre-operational test was carried out during the refueling outage, when the Emergency Diesel Generators "A" and "C" and Emergency Service Water Pump 46-2A were available.

This test procedure verified the operability of ESW Pump 46P-2B from the Control Room Panel 09-6, during normal operating conditions, and from Auxiliary Shutdown Panel 25ASP-3, in the event of Control Room evacuation.

The safety evaluation determined that the test was not described in the FSAR and concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-92-135

JAF-SE-92-092, Rev. 0 ALTERNATE SHUTDOWN CAPABILITY FOR SAFE SHUTDOWN COMPONENTS

The purpose of this nuclear safety evaluation was to examine the impact of modification F1-92-135 on plant safety. As a result of Appendix R validation and verification effort, it was determined that:

1. Both Reactor Head Vent Valves 02AOV-17 and 02AOV-18 could spuriously open in case of a fire in Fire Area 07 (Control Room). Also a fire in Fire Area 1C (West Cable Tunnel at 260') and 10 (Reactor Building at 272' West) could impair operation of both the valves.

2. Both Containment Spray Valves 10MOV-26B and 10MOV-31B could spuriously open in case of a fire in Fire Area 07 (Control Room).

Modification F1-92-135 provides the capability to isolate Reactor Vent Head Valve 02AOV-17 solenoid coil and isolation and control capability for Containment Spray Valve 10MOV-26B from the Control Room for a Control Room fire. An isolation switch and a control switch were added in Auxiliary Shutdown Panel 25ASP-3 for isolation and control of 10MOV-26B. This modification also provided re-routing of cables for Reactor Head Vent Valve 02AOV-18 through fire areas not accessed by cables for 02AOV-17.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: **TEST: POT-2K**

JAF-SE-92-093, Rev. 0 **ALTERNATE SHUTDOWN CAPABILITY FOR SAFE SHUTDOWN COMPONENTS**

The purpose of this nuclear safety evaluation was to examine the impact of pre-operational test (POT-2K) on plant safety. The test was conducted after the Reactor Vent Valve portion of the installation of modification F1-92-135 was completed and before commencement of the Reactor Vessel Operational Pressure Test (ST-39H). The test verified that the Reactor Head Vent valve 02AOV-17 can be isolated from the Control Room by operating isolation switch IS-1PCIN04 at the Remote Shutdown Panel 25RSP.

This Pre-Operational Test (POT-2K) for the Reactor Head Vent Valves was conducted during the refueling mode of the plant, when the reactor is depressurized in order to verify the operability of the valves.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A TEST: POT-02L

JAF-SE-92-095, Rev. 0 ISOLATION OF SRVs FOR APPENDIX 'R' FIRES

The purpose of this nuclear safety evaluation was to examine the impact of Pre-Operational Test POT-02L on plant safety. The test was conducted after the Safety Relief Valve portion of modification F1-92-078 was completed and before the refueling outage was over. The test verified that the eleven Safety Relief Valves (02RV2-71A, -71B, -71C, -71D, -71E, -71F, -71G, -71H, Panel 09-4 and can be isolated from the Control Room by operating isolation switches on the Auxiliary Shutdown Panel 25ASP-5 and bring the SRVs to a close position in the event of a Control Room fire. The test was conducted during the refueling mode of the plant, when the reactor is depressurized in order to verify the operability of the SRVs.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specifications or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-172

JAF-SE-92-096 **REMOVE 3/8" LLRT FITTINGS OUTBOARD OF ILRT CONNECTION ISOLATION VALVES AND ILRT OUTLET ISOLATION VALVES**

The purpose of this modification is to replace four (4) 3/8" ϕ tubing LLRT connections with 3/4" ϕ piping connections.

This modification removed 3/4" x 3/8" concentric reducers and associated tubing. Installed 3/4" pipe and threaded caps for local Leak Rate Test connections associated with containment isolation valves 14AOV-13A, -13B and 10AOV-68A, -68B. This modification was performed on tubing located outboard of 14CSP-402A, -402B and 10RHR-737A, -738B.

This modification involved the replacement of test connection tubing which is not safety-related plant equipment. The performance of this modification did not affect plant operation but only plant testing during times when the associated systems are declared inoperable.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A TEST: POT-33E

JAF-SE-92-097, Rev. 0 UPGRADE OF CST LEVEL INSTRUMENTATION LOOP POWER SUPPLY

The purpose of this pre-operational test was to verify that the Condensate Storage Tank Level Instrument Loop, 33LT-101/LI-101A which was modified by modification M1-92-069, will perform its required function as indicated in POT-33E.

The Condensate Storage Tank (CST) Level Instrument Loop was powered from a non-vital power source panel 71RRACB8. In the event of a loss of offsite power coincident with an Appendix R fire, CST level indication would have been lost. The powering of the CST level instrumentation loop from panel 71ACUPS-1 provides an uninterruptible power supply to the loop during the described scenario.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: **TEST: STP-93M**

JAF-SE-92-098, Rev. 0 **TEST OF EMERGENCY DIESEL GENERATOR OPERABILITY AND SUSCEPTIBILITY TO DC GROUNDS ON THE 125VDC DISTRIBUTION SYSTEM**

The purpose of special test STP-93M is to determine the effect that any credible ground or series of grounds on the 125 VDC distribution system has on the design basis operational requirements of the Emergency Diesel Generators. The 125 VDC distribution system supplies power to the engine speed controller on each EDG. The speed controller manufacturer had acknowledged that the model EGM used at JAF were susceptible to grounds on it's DC supply. Previous tests performed on the EDG's had shown that the diesels, when operating in parallel to the system, experienced oscillation when a low (unquantified) impedance path to ground was present on the 125 VDC supply. Therefore this special test provides data on EDG performance by observation of EDG operating parameters (speed, frequency, and KW) both with and without a ground present.

This safety evaluation reviewed the potential impact on plant safety of conducting a test on each Emergency Diesel Generator to determine their sensitivity to and the potential effects of a spurious ground appearing on the 125 VDC Distribution System.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-92-182

JAF-SE-92-099, Rev. 1 RELOCATION OF RELAY ROOM NORTH FIRE PROTECTION CO_2 PANEL FOR APPENDIX R FIRE*

The purpose of this modification was to relocate fire protection panel 76CO2-PNL-4, and associated pressure switch 76PS-138 from the Admin. Building Hallway to the Control Room Ventilation Room. Pushbutton 76PB-FPS-4 was removed and discarded. The associated manual dump valve was remained in the Admin. Building Hallway.

It can be concluded that the relocation of Panel 76CO2-PNL-4 and associated pressure switch did not degrade the design basis or functions of the plant equipment and was performed based on the following conclusions:

The modification did not degrade the qualification of any equipment. All fire protection guidelines were followed. New fire hazards were not created. There were no redundancy and separation criteria imposed on this modification.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-91-257

JAF-SE-92-100, Rev. 0 DRYWELL COOLING INLET DAMPER REMOVAL

The Drywell Cooling System is an air recirculation cooling system provided to remove and dissipate the Drywell heat gain. The original drywell cooling system design required 3 out of 4 cooling coils to be used under normal conditions. The inlet dampers were provided to isolate the spare coil. This design was changed with modification F1-75-017 to all 4 coils being used at all times. This eliminated the need for the inlet dampers. The removal of the cooling coil inlet dampers did not change the operation of the drywell coolers. The removal of the inlet dampers did aid in the replacement of the cooling coils and did remove a potential obstruction from the air flow path.

The removal of the inlet dampers of the A & B Drywell Cooler Assemblies did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety. There was no reduction in the margin of safety as defined by the design basis for any Technical Specifications as a result of this modification.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-91-205 TEST: POT-76AC

JAF-SE-92-101, Rev. 1 PRE-OPERATIONAL TEST (POT-76AC) FIRE PROTECTION, BATTERY ROOM CORRIDOR BR5

The purpose of this nuclear safety evaluation was to examine the impact of Pre-Operational Test (POT-76AC) on plant safety. The test was conducted to verify the new automatic water fire suppression system that was installed in the Battery Room Corridor (Fire Zone BR5) by Modification M1-91-205.

This pre-operational test procedure POT-76AC was conducted during the cold condition when the reactor was depressurized. Pre-operational test POT-76AC verifies the installation and operability of the new cables and annunciator signals associated with this system. This test procedure did not conflict with or degrade the design basis or function of any other plant system or components and hence, it was performed without compromising the safety of the Plant.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specifications or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION:

JAF-SE-92-104 A CONTINGENCY PLAN TO SUPPLY COOLING WATER TO EDG "A" OR "D"

The purpose of this temporary modification was to provide an alternative cooling water source and independent water discharge line for the EDG "D" Jacket Water Cooler (93WE-1D) and EDG "A" Jacket Water Cooler (93WE-1A). This temporary modification installed 4 inch fire hose inlet and outlet attachments to the tube side of the A and D EDG Jacket Water Coolers by replacing the existing blind end plate with a temporary end plate with inlet/outlet nozzles. An independent supply and return water system was installed for emergency diesel generator use.

During portions of implementation of Modification F1-92-068, both trains of Emergency Service Water were isolated. The plant was in the cold condition and in accordance with the Technical Specifications as discussed in Reference 1.

The plant is supplied from the 345kV and both 115kV sources of offsite power. However, in the unlikely event of loss of all sources of offsite power, one of the two (EDG "A" or "D") diesel generators was made available to supply the required loads.

The safety evaluation was performed to support the Temporary Modification, which supplies cooling water to either Emergency Diesel Generator Jacket Water Cooler "A" or "D" while emergency service water is unavailable. This evaluation results in no unreviewed safety questions.

The safety evaluation concluded that the temporary modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION:

JAF-SE-92-105, Rev. 1 TEMPORARY MODIFICATION TO BYPASS AUTO-START CAPABILITY OF EDG'S, ESW PUMPS AND MOV'S

This Temporary Operating Procedure was to render the ESW pump automatic start, ESW and RBCLC system MOVs automatic repositioning on RBCLC low pressure signal inoperable to support the temporary modification to install scaffolding and extend existing EDG Jacket Cooler Return Line 46-12"-WES-151-9 below water level in ESW Pump Bay "A".

The operating procedure was temporary and performed during a plant shutdown condition. The EDG inoperability during a refueling outage had been reviewed and the plant was within the Technical Specifications.

The EDG and ESW inoperability during a refueling outage had been reviewed and the plant was within the Technical Specifications.

The plant was in a cold condition in accordance with the Technical Specifications and all equipment was available for manual operation.

This Temporary Operating Procedure did not affect the Security Plan. The Appendix R requirements were not impacted. The Quality Assurance Program was not degraded because the Temporary Operating Procedure was strictly in accordance with 10CFR50, Appendix B.

The environmental parameters did not change as a result of this Temporary Operating Procedure.

MODIFICATION:

JAF-SE-92-107, Rev. 0 CONCRETE MASONRY WALLS IN CLASS I BUILDINGS

The purpose was to evaluate the structural integrity of unreinforced concrete masonry walls in Class I buildings. This evaluation considered the discrepancies found between the as-built configuration and what is stated in FSAR Section 12.4.6.1 for concrete masonry block walls.

FSAR Section 12.4.6.1 states that: "...all concrete block walls in seismic Class I structures are constructed with hardware cloth as horizontal joint reinforcement on top of the first course and at every third succeeding course in every interior wall. Dowels from concrete slabs are provided for lateral support at the base". Contrary to this statement, the block wall in front of 10MOV-17 at elevation 272′ of the Reactor Building (as stated in AQCR 92-017) did not contain both the hardware cloth and the dowels.

The review of existing documentation showed that all safety-related masonry walls were evaluated by using as-built conditions for the IE Bulletin 80-11 program. Stone & Webster surveys performed during this re-evaluation program documented whether hardware cloth and dowels existed for each wall. Subsequently, each wall was analyzed taking into consideration the existence and absence of the hardware cloth and dowels. All safety-related masonry walls were requalified in accordance with IE Bulletin 80-11 requirements. This evaluation was submitted and reviewed by the NRC.

The safety evaluation concluded that the evaluation did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-212

JAF-SE-92-108, Rev. 0 UPGRADE THE PERFORMANCE CAPABILITY OF RWCU RETURN CONTAINMENT ISOLATION VALVE OPERATOR

The purpose of this modification was to upgrade the thrust capability of 12MOV-69 from 5030 pounds of thrust to 9852 pounds of thrust at the recommended torque switch setting to comply with NRC Generic Letter 89-10. This document states that the valve friction factor should be based upon conservative test reports as opposed to non-conservative assumptions.

The motor was changed from a 5 ft-lbf torque motor to a 15 ft-lbf torque motor. The spring pack was changed from a 0301-111 (55 to 145 ft-lbf) to a model 0301-113 (170 to 240 ft-lbf) and the SB operator compensator spring was changed from a "light thrust assembly" to a "heavy thrust assembly".

The new valve operator parts did not cause plant accidents as described in chapter 14 of the FSAR or other new accidents since the parts are standard Limitorque parts that are suitable for the application and the valve has been evaluated for the higher thrust and torque loads applied concurrently with seismic loads.

The modified valve operator has increased thrust capability and is more capable of performing its intended safety function (reference 12.1).

The additional valve weight did not require reanalysis to establish seismic qualification since the valve replacement analysis by Cigna (F1-87-068) used an enveloping value for valve weight.

The valve operator is rated for 14,000 lbf thrust which is the thrust generated at the new maximum torque switch setting. Therefore, the existing operator parts are qualified for the new thrust and torque loads.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-208

JAF-SE-92-109, Rev. 0 CSP/RHR CHECK VALVE TEST CONNECTIONS

The purpose of this modification is to provide a means to facilitate reverse flow testing of the existing Core Spray condensate transfer discharge check valves and Residual Heat Removal condensate transfer discharge check valves listed above. A small section of branch piping with valve shall be installed to facilitate the test. These connections have provided plant personnel the ability to verify the operability of the individual safety-related boundary check valves for In-Service Testing, as required by NRC/SER on revision 4 of the second interval IST program and has allowed the exercising and flushing of corrosion products from the valve internals.

This modification was achieved by installing a branch piping containing a 3/4" ϕ VGS-60B gate valve with 3/4" diameter nipples and threaded cap onto the applicable piping section. This connection was accomplished by using reinforced welded connections on the 4" diameter RHR piping and the 2" diameter core spray piping. In addition, check valves which are located on piping with insufficient space to allow the installation of a test connection were relocated. This relocation will not change the original design function of the applicable check valve and will maintain the piping system integrity.

There will be no changes to the design function of the CSP/RHR piping systems as a result of adding the check valve test connections. This modification does not affect postulated radiological releases, since the valve on test connection will remain shut during normal plant operation. Therefore, has no affect on the QA, Fire Protection, Security, or EQ Programs. FSAR change noted in Section 7.0, effect on Final Safety Analyses Report does not alter or reduce margin of safety.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-214

JAF-SE-92-111, Rev. 0 HPCI\RCIC TORUS SUCTION VALVES, AND CORE SPRAY OUTBOARD INJECTION VALVES REMOTE MANUAL CLOSURE CAPABILITY.

The purpose of this minor modification was to change the valve control circuitry for each of the High Pressure Coolant Injection (HPCI) 23MOV-57 and 58 and Reactor Core Isolation Cooling (RCIC) 13MOV-39 and 41 torus suction valves. The reason for the valve control circuitry change is to enable these valves to be closed by remote-manual operation from the control room should primary containment isolation be required, even though automatic transfer logic signals are present that require these valves to be open.

The design change to the HPCI and RCIC torus suction valve is designed to minimize the plant operator interface with the automatic transfer function for either the HPCI or RCIC system. Minimal operator interface and testing ensures each system operation and the additional feature capability to remotely close these valve under any accident condition to ensure that primary containment can be obtained. Therefore, the proposed minor modification does not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety, does not create the possibility of an accident or malfunction of a different type, does not reduce the margin of safety as defined in the basis for any Technical Specification and does not involve an unreviewed safety question.

MODIFICATION: F1-92-178

JAF-SE-92-113, Rev. 3 INSTALLATION OF EMERGENCY LIGHTING UNITS

The purpose of this minor modification was to resolve emergency lighting inadequacies discovered during an NRC Fire Protection/Appendix R inspection and a revision to the Appendix R Analysis (Ref. T, Ref. U). This modification is a QA Category M modification. It was determined during testing that existing emergency lighting for various areas requiring operator access/egress or operator actions associated with safe shutdown equipment in accordance with 10CFR50 Appendix R Section III.J are inadequate. The areas covered within the scope of this modification are the Heater Bay, Electrical Bay, Admin. Building, Emergency Diesel Generator Building, Relay Room, and Reactor Building. Future modifications will cover additional areas.

Exide F-100 and Holophane M-19 Emergency Lighting Battery/Charger packs are connected to existing normal light circuits to provide additional lighting in the required areas. This configuration will provide emergency lighting for a minimum of 8 hours continuous operation upon the loss of normal lighting power.

It can be concluded that the modification of the Emergency Lighting has not degraded the design basis or functions of the plant equipment and can be performed based on the following conclusions:

Modification F1-92-178 has enhanced illumination for various areas requiring operator access/egress and equipment operation.

The modification did not degrade the qualification of any existing components. All fire protection guidelines are being followed. New fire hazards were not created. Additional combustibles (cable were not added.

It has met all design and material requirements for a QA Category M installation.

The modification has no impact to the Security Plan. The modification does meet QA Category M criteria and will meet the Fire Protection System requirements.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-192

JAF-SE-92-114, Rev. 0 ADDITION OF CO_2 WARNING LIGHTS AND SIGNS FOR HVAC FAN ROOMS AND SIGNS FOR CABLE RUN ROOMS

This modification installed four CO_2 system actuated alarm bells and strobe lights (76AB-20, 21, 22, 23) in the Turbine Building HVAC fan rooms located at elevation 286'-0" above the East and West Electric Bays, Fire Areas 1C and 2 respectively. In addition, appropriate signs to instruct personnel to exist these areas as well as the North and South Cable Run Rooms were added.

This modification ensured personnel warning and provide appropriate direction to personnel in the areas in the event of a CO_2 Fire Protection System discharge. No changes to the Technical Specifications or the FSAR were required as a result of this modification. Background pertinent to this modification is described in Memorandum NED-92-RK-081, dated April 28, 1992.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-207

JAF-SE-92-115, Rev. 0 REMOVAL OF STANDBY LIQUID CONTROL SYSTEM PUMP DISCHARGE ACCUMULATIONS

The Standby Liquid Control (SLC) pump discharge accumulators (11ACC-7A/B) were designed to dampen high pressure hydraulic pulsations generated at the discharge of the positive displacement pumps. The internal bladder assembly of each accumulator is charged with 450 psig nitrogen.

Every time the Standby Liquid Control (SLC pumps are operated for surveillance testing, the pressure of the discharge accumulators is verified. To perform the verification, a pressure gauge is connected to a Schrader fitting at the top of each accumulator to measure the internal nitrogen pressure. Due to a very slow leakage past the charging connector spring loaded valves (Schrader fittings), a loss of nitrogen pre-charge in the accumulators occurs. In an attempt to correct this problem a "Schrader fitting to pressure gauge adapter" was installed. Unfortunately, the new adapter has not resolved the leakage past the Schrader fitting. The loss of nitrogen pre-charge has resulted in repeated instances of SLC system outages and the initiation of LER-91-013.

The installation of a modification to remove the SLC pump discharge accumulators and install test valves to facilitate in-service testing would delete equipment that is not required to support the operability of the SLC system. This would also clarify the required configuration of the system. The in-service testing involves the testing of SLC Pump Discharge Check Valves 11SLC-43A and 11SLC-43B in accordance with IST procedure ST-6H (Rev. V-18). This test procedure will be revised at a later date to incorporate the changes to the SLC system performed by this Minor Mod.

The removal of the SLC accumulators in accordance with the proposed design and plant procedures did not pose a safety concern since the SLC system has been designed to preclude the adverse affects of system pressure spikes resulting form the use of positive displacement pumps.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-92-180

JAF-SE-92-116, Rev. 1 MODIFICATION TO WATER CURTAIN TO ELIMINATE OBSTRUCTIONS

The purpose of this modification was to extend the existing water curtain coverage in the reactor building on elevations 227'-6" (East Crescent, Curtain 1), 272'0" (South, Curtain 2) and 300'0" (West, Curtain 3) to resolve equipment/pipe obstructions greater than 24 inches.

The modifications to the subject water curtains is an improvement to the existing plant fire protection system and did not increase the probability of occurrence o consequence of an accident or malfunction of structures, systems, or components important to safety previously evaluated in the FSAR. In addition, the modifications to the water curtains did not create the possibility of an accident or malfunction of a different type than evaluated in the FSAR. The flooding analysis with respect to the worst case water spray accident was addressed to include the additional nozzles associated with the water curtain extensions. The analysis concludes that the extensions to the water curtain did not affect the safe shutdown capabilities of the plant.

The NRC granted NYPA an exemption request, Reference 2.0, to use the water curtain system as Appendix R Barriers. The modifications to the water curtains are being made to assure compliance with the water curtain requirements specified in Reference 1.0.

The modification is being designed to QA Category M, seismic Class II criteria as the existing system and as previously approved. Since the modification is an improvement to the fire protection system, the fire protection system was not degraded.

Upon activation of the worst case fire spray accident, approximately 12 gpm would be added to the flooding in the reactor building. This water, per Reference 5.11, would be contained in the East and West Crescent Areas and therefore did not have an impact on the environment.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-209

JAF-SE-92-117, Rev. 0 TORQUE SWITCH REWIRING FOR MOV MOTOR PROTECTION

This modification rewired the torque switches for the motor operated valves listed in Section 1.0. These motor operated valves are required to operate for alternate safe shutdown of the plant. The torque switches were rewired to place them between the motor control centers and the control room/alternate shutdown panel. This modification prevented the bypassing of the torque and associated limit switches by hot shorts in the valve control circuitry due to a fire in the control room. A hot short could result in sustained high MOV motor current prior to the operator shifting valve control to the alternate shutdown panel. This action plan was presented in NRC Information Notice 92-18 based on the concern that a control room fire could cause hot shorts in the valve control circuitry resulting in damage to the MOV if the motor thermal overloads were bypassed. The JAF thermal overloads were not bypassed, however a concern exists that they may not be appropriately sized. An evaluation of the thermal overload sizing is being performed as a separate modification independent of the Fire Protection Appendix R restart issues. The rewiring of the torque switch for those MOVs which are required to operate for alternate safe shutdown of the plant, will address the valve operation concerns associated with Appendix requirements.

In addition to the torque switch rewiring, seal in circuitry is being added to 10MOV-70A and 10MOV-70B so that valves are in accordance with their design basis. This proposed minor modification does not design basis or function of any plant equipment, and can be performed without comprise to safety.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-210

JAF-SE-92-118, Rev. 0 EMERGENCY DIESEL GENERATOR SPEED CONTROLLER 125VDC - 48VDC ISOLATED POWER SUPPLY INSTALLATION

This modification installed a 125VDC - 48VDC isolated power supply in each of the emergency diesel generator control panels 93EGP-A, 93EPG-B, 93EGP-C and 93EGP-D located in the switchgear room of the diesel generator building. The new isolated power supply is intended to isolate the existing EG-M speed controller units for emergency diesel generators A, B, C, and D (System 93) from the 125VDC power system (System 71). NYPA memorandum to technical services dated 4/23/92 addressed swings in the emergency diesel generator output caused by grounds on the 125VDC system. The grounds cause a fluctuation in the DC voltages between the positive bus, negative bus and station ground. The EG-M speed controller is susceptible to these ground disturbances and will respond, causing the actuator to oscillate. The disturbance is transient in nature and lasts a fraction of a second as long as the ground is fixed in value. Should the ground vary in magnitude, the generator's output oscillation may become unacceptable and could lead to an emergency diesel generator trip. The installation of the power supply was intended to isolate the EG-M speed controller from the voltage fluctuation caused by the grounds by providing a regulated source of 48VDC.

This modification installed isolated power supplies which provide a regulated source of 48VDC for each EG-M control box with an input of 110VDC-370VDC. The isolated power supplies are qualified to provide rated power to the EG-M control box while operating with an input voltage as low as 90VDC. (see Attachment 13.8). This installation ensured that the EG-M control boxes function as intended, thereby providing increased assurance that proper control of the speed and loading of the emergency diesel generators is maintained. The power supplies are commercial grade and have been qualified by Farwell & Hendricks as a dedicated item. This modification did not present an unreviewed safety question, did not degrade the design basis or function of any plant equipment, and were performed without compromise to safety.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-229

JAF-SE-92-119, Rev. 0 **INSTALLATION OF PIPE RUPTURE RESTRAINT ON RCIC LINE 13-3"-SHP-902-17 REACTOR BLDG - TORUS ROOM**

A pipe rupture restraint was required to be installed at the RCIC steam supply line to prevent impact to the RCIC vacuum breaker line in the event of a high energy line break. Piping that cannot be isolated from the containment should be protected from the effects of the pipe whip and jet impingement.

The location of the rupture restraint is determined based on the NRC's Branch Technical Position MEB 3.1.

In order to prevent damage from the effects of a postulated pipe break in the 3" RCIC steam line acting on the 1½" RCIC vacuum breaker line, a pipe rupture restraint is installed on the 3" RCIC line.

Upon completion of this modification the RCIC piping system will be in the same operable status as in the original configuration. Therefore, this modification does not change the existing plant design basis.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: **TEST: POT-10AB**

JAF-SE-92-120, Rev. 0 **ALTERNATE SHUTDOWN CAPABILITY FOR SAFE SHUTDOWN COMPONENTS**

The purpose of this nuclear safety evaluation was to examine the impact of Pre-Operational Test (POT-10AB) on plant safety. The test was conducted to verify that Containment Spray Outboard valve 10MOV-26B functions as intended after implementation of the modification F1-92-135. The modification provides additional control capability for this valve from Auxiliary Shutdown Panel 25ASP-3 in the event of a control room fire. The test verifies the operability of the valve from the Main Control Room Panel 09-3 and from Auxiliary Shutdown Panel 25ASP-3. The test was conducted when Installation Procedure F1-92-132 IP #1 for valve 10MOV-26B was completed. The installation and test were done during 1992 Refueling Outage, when the reactor was in the cold condition and depressurized.

This test procedure did not conflict with or degrade the design basis or function of any other plant system or components and hence it was performed without compromising the safety of the plant.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specifications or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-149

JAF-SE-92-121, Rev. 0 INCREASED STROKE TIME FOR RHR CONTAINMENT SPRAY OUTBOARD ISOLATION VALVES

The purpose of modification M1-92-149 was to increase the operator thrust for valves 10MOV-26A and B in order to meet NRC Generic Letter 89-10 MOV sizing criteria. Valve Operator thrust was increased by regearing the motor operator. Motor operator regearing also lengthen the stroke time from ten seconds to twenty seconds. A longer stroke time is technically acceptable since the operating time of one minute for these valves has been analyzed by General Electric and found acceptable (ref. 12.9, 12.10 and 12.11). Valves 10MOV-31A and B are in line with 10MOV-26A and B, and the plant licensing commitments for these valves were also changed to reflect the one minute acceptance criteria. Valves 10MOV-31A and B were not changed physically with this modification.

FSAR sections 6.4, 7.3 and 14.6 have been reviewed and will be revised to include the results of these analysis.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-90-202

JAF-SE-92-122, Rev. 0 RWCU PUMP "A" SIDE PROCUREMENT AND INSTALLATION

The purpose of this modification was to remove the existing non-functioning "A" side Reactor Water Cleanup (RWCU) Pump, 12P-1A, and install in its place, a replacement pump, similar to that installed and operating on the "B" side. The replacement pump is a 50% system design flow pump, thereby restoring the RWCU system to 100% of its design flow capacity, for parallel pump operation.

The pump (and its drive motor) have been purchased to specifications which meet or exceed the original specification requirements. The pump function and design will not be changed by this modification, and, therefore, the operation of the Reactor Water Cleanup System will not be negatively impacted.

Based on the review performed herein, the following is concluded for the Safety Evaluation:

The original design basis has not been changed.

The replacement pump was purchased to specifications that met or exceed the original specification requirements. The piping and supports have been designed in accordance with the appropriate codes and standards and the piping maintained its pressure boundary under all conditions of operation.

This modification did not affect the Technical Specifications or the safety analysis of the plant as described in the JAFNPP FSAR, Chapter 14.

System operation has not changed in any way.

This modification did not affect the Fire Protection, it did not impact any safe shutdown components or fire barriers and it did not increase the combustible levels in any of the plant areas. This modification was implemented in accordance with the site's Quality Assurance program.

The proposed modification did not affect the environmental impact of the plant or involve an unreviewed environmental question, because this modification did not discharge any effluents.

MODIFICATION: F1-91-355

JAF-SE-92-123, Rev. 2 THREE HOUR FIRE SEALS FOR BONDSTRAND PIPING

The purpose of this modification was to modify or remove the fiberglass (Trade names Bondstrand and Greenthread) and PVC piping in twenty two (22) existing non-qualified penetration seals listed in Table A of Ref. 5.1. Drain traps were added to three (3) fiberglass pipes and one drain trap for a sink drain line was removed, all located above penetrations at floor el. 272'0". The drain line for one fume hood sink in the Chemistry Laboratory was retired, and the pipe internally sealed with silicone elastomer within the penetration. Fiberglass and PVC pipe in fourteen (14) penetrations were replaced with steel or alloy piping spool pieces, and the penetrations resealed with an appropriate thickness of RTV silicone foam or silicone elastomer. Several of these spool pieces were wrapped with an endothermic material. This modification resulted in a 3-hour fire rating for each of these penetrations.

One additional penetration (S-4024), containing 1" PVC pipe, is not presently sealed. Stainless steel tubing (3/8" dia.) has replaced the through penetrating PVC pipe to a minimum of 5 feet on each side of the wall. The penetration will be sealed with an appropriate thickness of silicone elastomer, resulting in a 3-hour fire rating.

This modification is Category M for reconfiguration of penetration seals. All penetrating piping is Category II/III.

This modification has been implemented to comply with 10CFR50, Appendix R and BTP APCSB 9.5-1, Appendix A, which require 3-hour rated seals at these penetrations.

It is concluded that this modification did not degrade the designed safety or function of the plant, based on the following:

The systems being modified are not safety-related and perform no safety functions. One Secondary Containment penetration (S-329) was opened and resealed to its original condition (with the original thickness of silicone foam).

The modification did not degrade existing systems, structures and components. The functions of the systems affected by this modification are not addressed in the Technical Specifications. One Secondary Containment penetration (S-329) was opened and resealed to its original condition (with the original thickness of silicone foam).

This modification did not interface with the Security System, the site's Quality Assurance Program will be adhered to, and the level of fire protection will be enhanced by the resultant piping configuration and qualified seal arrangement.

MODIFICATION: M1-92-231

JAF-SE-92-124, Rev. 0 DELETE TIME METER FOR TANK VENTS EXHAUST FAN 66 FN-35

The elapsed time meter in the 71MCC-141-OC3 was disconnected by the temporary modification (jumper) 90-012 in January 1990. This modification removed and document removal of the elapsed time meter.

Removal of the elapsed time meter had no affect on the function of circuit breaker of 71MCC-141-OC3. Implementation of this modification to remove elapsed time meter did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety. This modification did not create the possibility of an accident or malfunction of a different type than those evaluated in JAF FSAR. There is no reduction in the margin of safety as a result of this modification. The proper QA requirements have been met for the preparation of documents, installation and procurement of materials for this modification.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-91-252

JAF-SE-92-125, Rev. 0 CYCLE 11 RELOAD CORE

This evaluation documents safety analyses performed on the Cycle 11 core and the thermal limits established in the Cycle 11 Core Operating Limits Report (COLR). The core was loaded with 152 fresh GE11 fuel bundles. Cycle 11 specific analyses were performed according to the methodologies detailed in Reference 1* and these analyses demonstrate that the core can be operated safely within the limits specified in the COLR. This evaluation also covers activities related to reloading the core from the offloaded condition to the planned Cycle 11 configuration.

The refueling procedure has been reviewed for potential reactivity issues. The safety analyses performed on the Cycle 11 core have been reviewed and found adequate in scope and content against the requirements in Reference 1* and the FitzPatrick FSAR and Technical Specifications. Fuel design changes have been reviewed against designs previously used at FitzPatrick. On the basis of this review no unreviewed safety question exists, and the safety analyses properly support the limits established in the Core Operating Limits Report (COLR).

The probability of occurrence of an accident or abnormal operational transient (occurrence) is unaffected by the changes described in this evaluation since the plant and its operation is not changed; no accident or abnormal occurrence analyzed in the FSAR initiates in the fuel itself. The consequences of an accident or abnormal operational transient (occurrence) are unaffected since the analysis methods and procedures ensure that unacceptable safety results will not occur. Fuel operating limits, i.e. MCPR, MAPLHGR, and MLHGR are determined by the analyses such that the unacceptable safety results as detailed in FSAR Sections 14.2 and 14.3 and Reference 1* will not occur. The core loading pattern and fuel reactivity are determined based on the ability to meet Technical Specification shutdown requirements (Section 3.3) and FSAR Section 14.6 and Reference 1* accident limits. The review of the fuel design changes incorporated into Reload 10 have not identified any new type of malfunction or accident not previously considered in the FSAR or Reference 1*. The margin of safety defined in the Technical Specifications for fuel thermal limits and shutdown margin are unaffected by the changes described in this evaluation. Margins of safety are provided by the conservative assumptions of fuel behavior i.e. Appendix K limits for loss of coolant accident analysis or transition boiling as criterion for fuel damage during transients. This conservatism is further ensured by using conservative calculational models. Cycle and fuel design specific thermal limits are obtained by application of these models; the margin of safety is preserved by adjustment of the limits such that the consequences of analyzed events are bounded by FSAR and/or Reference 1* values.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

* Reference 1 - GE Report, General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A-10, February 1991.

MODIFICATION:

JAF-SE-92-126, Rev. 0 DOOR FAN PRESSURE TEST - RELAY ROOM (SPECIAL TEST STP-76AD)

The door fan pressure test was performed to determine the integrity of the Relay Room as an enclosure envelope to retain a delivered concentration of CO_2 for an NFPA 12 recommended soak time. The testing method provided in NFPA 12A-1989 Appendix B was used to collect the data required to mathematically approximate the maximum hold time.

The performance of this test did not include the discharge of CO_2 in the room. The test also determined the integrity of the seals in the Relay Room by ensuring that a positive pressure is maintained inside the Relay Room pressure boundary.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-92-173

JAF-SE-92-128, Rev. 1 RCIC VALVE CONTROL CIRCUIT MODIFICATION

The purpose of this modification was to reroute a cable and modify circuits associated with the RCIC Control System. This is necessary to ensure that a fire will not disable RCIC Control, prevent the operation of the RCIC Speed Controller (13FIC-91) or prevent the ability to isolate the RCIC steamline or operate the RCIC turbine. In addition, a RCIC Manual Initiation Switch is provided to ensure RCIC initiation when RPV low level logic signals are lost due to a fire. This modification was implemented to comply with the requirements of 10CFR50, Appendix R. This is a QA Cat. I modification.

It can be concluded that the RCIC control circuit modifications did not degrade the design basis or functions of the plant equipment and can be performed based on the following conclusions:

This modification upgraded the RCIC system to be capable of RCIC steam line isolation and RCIC manual initiation for plant fires in compliance with 10CFR50, Appendix R criteria.

This modification has met all design and material requirements for a QA Category I installation.

The modification has no impact to the Security Plan. The modification has met QA Category I criteria and has met the Fire Protection System requirements.

There are no new releases, release passways or chemical concerns associated with this modification.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-209 TEST: POT-MISC-7A

JAF-SE-92-130, Rev. 0 PRE-OPERATIONAL TEST FOR TORQUE SWITCH REWIRING MODIFICATION M1-92-209

Pre-operational test POT-MISC-7A was performed to check the operability of 23 Motor Operated Valves (MOVs) (listed below) which have had their torque and limit switches re-wired or have been modified by modification M1-92-209.

The MOVs torque and limit switches were rewired in order to prevent them from being bypassed by hot short due to a fire in the control room as described in NRC Information Notice 92-18. A hot short could have caused sustained high MOV motor current prior to the operator shifting valve control to the alternate shutdown panel. The torque and limit switches are now wired between the motor control center and control room/alternate shutdown panel.

Modification M1-92-209 also added seal-in circuitry to 10MOV-70A & 10MOV-70B so that they would be in accordance with their design basis. 10MOV-70B is also one of the 22 MOVs which have had their torque and limit switches rewired.

10MOV-12B	10MOV-39B	12MOV-18
10MOV-13D	10MOV-65B	23MOV-16
10MOV-15D	10MOV-66B	23MOV-25
10MOV-16B	10MOV-70A	23MOV-60
10MOV-21B	10MOV-70B	29MOV-77
10MOV-25B	10MOV-89B	46MOV-101B
10MOV-26B	10MOV-148B	46MOV-102B
10MOV-27B	10MOV-149B	

The testing of the modified MOVs did not change the original designed operation/function of the MOVs.

The testing of the modified MOVs did not change the safety function of the MOVs.

The testing of the modified MOVs did conform to the existing Quality Assurance Program and did not affect the Security Plan or Fire Protection System.

Functional testing as called out in section 9 of POT-MISC-7A was performed prior to declaring a MOV modified by modification M1-92-209 operable.

POT-MISC-7A was completed in its entirety prior to unit start-up.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-92-173 TEST: POT-13F

JAF-SE-92-131, Rev. 0 PRE-OPERATIONAL TEST FOR RCIC VALVE CONTROL CIRCUITS FOR MODIFICATION F1-92-173

Pre-operational test POT-13F was performed to verify the operability of RCIC Isolation Valve 13MOV-16 and RCIC Trip and Throttle valve 13HOV-1, which have had their valve control circuits modified by modification F1-92-173. This preoperational test also verifies the operability of fuses 13A-F34 and 13A-F35 and relay 13A-K42-2, added for the Condensate Storage Tank "A" and "B" Low Water level RCIC control circuit, pushbutton switch "RCIC MANUAL INITIATION 13A-S18A" which provides an additional mean for initiating RCIC from the control room. This Preoperational test is QA Category I.

It can be concluded that POT-13F did not degrade the design basis of functions of the plant equipment and can be performed based on the following:

The testing of the modified valve control circuits did not change the original designed operation/function of the valves. RCIC turbine operation and primary containment is not required.

The testing of the modified valve circuits did not violate primary containment.

The testing of the modified valves have met QA Category I criteria and did not affect the Security Plan or Fire Protection System.

There are no new releases, release passways or chemical concerns associated with POT-13F.

It is concluded that this modification does not involve an unreviewed safety question. This is based on a review of the proposed preoperational test, FSAR updates and technical specifications.

MODIFICATION:

JAF-SE-92-133, Rev. 0 CHEMICAL TREATMENT OF CIRCULATING WATER SYSTEM TO REMOVE ZEBRA MUSSELS

The purpose of this treatment program was to remove zebra mussels from the circulating water system by adding a molluscicide (Betz Clamtrol CT-1) to the intake cap with one circulating water pump and normal service water in operation during the treatment time frame. Detoxification of the molluscicide was required and was accomplished through the addition of a detoxification agent in the accessible portion of the discharge tunnel in the screenhouse area. This safety evaluation addresses the effects of this chemical treatment procedure on safety related equipment and systems and the availability of the ultimate heat sink during and after the completion of the chemical treatment process.

In the event of the loss of station air, used to power the detoxification pumps, a back-up source of compressed air was onsite to ensure that detoxification of the treated water will take place.

The use of CT-1 product required a modification of the JAF SPDES permit. The NYSDEC approved the JAF request for the use of this product (Reference 20). JAF was required to deactivate the CT-1 chemical prior to discharging any water to Lake Ontario. JAF deactivated the CT-1 chemical with the Betz DT-S chemical slurry. By complying with the NYSDEC requirements, there was no detrimental effects to the environment.

This safety evaluation also considered the corrosive effects of the chemical on system materials equipment, the effect on plant equipment from "sloughage" of zebra mussels, and the compatibility of the chemicals with the fire protection plan, Control Room habitability, storage and spill control, and health effects on personnel handling the chemicals. None of these issues were found to be compromised by the use of CT-1 at JAF.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-92-166 TEST: POT-23K

JAF-SE-92-134, Rev. 0 PRE-OPERATIONAL TEST (POT-23K) FOR HPCI VALVE CIRCUITS FOR MOD F1-92-166

Preoperational test POT-23K was performed to verify the design function (to open and close during accident conditions) operability of HPCI Isolation Valves 23MOV-15 and 23MOV-16 and HPCI turbine stop valve 23HOV-1, which have been modified by Modification F1-92-166.

The proposed pre-operational test:

The testing of the modified valves did not change the original designed operation/function of the valves. HPCI turbine operation and primary containment are not required.

The testing of the modified valves did not violate primary containment.

The testing of the modified valves did conform to the existing Quality Assurance Program and did not affect the Security Plan or Fire Protection System.

Preoperational testing was completed prior to the next unit start-up.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-255

JAF-SE-92-135, Rev. 0 HPCI/RCIC SPEED CONTROLLER 125VDC/48VDC ISOLATED POWER SUPPLY INSTALLATION

This modification installs a 125VDC to 48VDC isolated power supply in the 125VDC power circuit for the HPCI and RCIC turbine EG-M speed control systems. The new isolated power supplies are intended to isolate the EG-M speed controller units from grounds on the 125VDC system. The EG-M speed controllers for the HPCI and RCIC turbines are the same as those utilized in the Emergency Diesel Generators (EDG's) speed control system. Recent testing on the EDG's has demonstrated that the EG-M speed controller unit is susceptible to grounds on the 125VDC system which powers it, and has resulted in EDG output oscillations. Due to the similarities with the EDG, HPCI and RCIC EG-M speed control systems, a concern exists that the HPCI and RCIC turbine operation may be suspect when grounds are applied to the 125VDC system.

This modification replaced the existing voltage dropping resistor assembly with a 125VDC to 48VDC isolated power supply thereby isolating grounds on the 125VDC system from the 48VDC required by the EG-M speed controller unit. Therefore, this modification enhanced the operation of the HPCI and RCIC turbines and ensure that output oscillations do not occur by providing the EG-M speed controller units with a clean (regulated) supply of 48VDC power.

The isolated power supplies were qualified to provide rated power to the EG-M control box while operating with an input voltage as low as 90VDC. This installation ensured that the EG-M control boxes function as intended, thereby providing increased assurance that proper control of the speed of the HPCI and RCIC turbines is maintained. No changes to the technical specifications or the FSAR were required as a result of this minor modification. The power supplies are commercial grade and were qualified by Farwell & Hendricks. The power supplies were procured form Farwell & Hendricks as a dedicated item. This modification did not present an unreviewed safety question, did not degrade the design basis or function of any plant equipment, and was performed without compromise to safety.

MODIFICATION:

JAF-SE-92-136, Rev. 0 POST WORK AND POST MODIFICATION PRESSURE AND LEAK TEST REQUIREMENTS

This safety evaluation analyzed changes to the pressure and leak test program. These changes provide pressure and leak test requirements to be used following maintenance, repair, replacement and modification of existing plant fluid systems and following installation of new plant fluid systems. These changes did not apply to regularly scheduled ISI tests, reactor vessel tests, or tests required by plant technical specifications.

The changes to the post-work and post-modification pressure and leak test program removed ambiguity associated with the existing program. The changes are consistent with applicable design codes and standards and incorporate a test methodology that is equivalent with the existing test program. Also, plant Technical Specifications did not apply to the changes, and the changes did not degrade the Security Plan or Fire Protection Systems. Based on the above, the changes were considered to involve no unreviewed safety question.

MODIFICATION: M1-92-244

JAF-SE-92-138, Rev. 1 ALTERNATE APPENDIX R REACTOR VESSEL LEVEL INSTRUMENTATION

This modification installed one alternate reactor vessel level transmitter (02-3LT-94) and one local level indicator (02-3LI-93) on Rack 25-51 at elevation 272'0" (Column R Line 3) of the Reactor Building.

A fire in Fire Areas 08 or 09 of the Reactor Building may impact both redundant Division A and Division B reactor vessel level transmitters. The fire has the potential of boiling the water contained in the instrumentation tubing located in the subject fire areas resulting in erroneous indication in the Control Room. To ensure availability of reactor vessel level indication, an alternate level transmitter and level indicator was installed on Rack 25-51 (Fire Area 10).

This modification ensured availability of reactor vessel level indication, in the event of a fire in Fire Areas 08 and 09 of the Reactor Building. All instrument tubing, from penetration X-28Aa and X-40 is contained within Fire Area 10. No changes to the Technical Specification were required as a result of this modification. An FSAR change has been submitted.

This modification did not present an unreviewed safety question, did not degrade the design basis or function of any plant equipment and was performed without compromise to safety.

MODIFICATION:

JAF-SE-92-140, Rev. 0 **SAFETY EVALUATION TO SUPPORT THE PERFORMANCE OF TOP-43A, BATTERY 'A' CONTROL BOARD (71BCB-2A) OUTAGE AND RESTORATION**

This Safety Evaluation documented the shutdown risk assessment and safety significance of removing the "A" battery control board, 71BCB-2A, from service to facilitate breaker maintenance. TOP-43A was utilized to control plant configuration while 71BCB-2A was out-of-service.

Since the plant remained in cold condition throughout the bus outage the margin of safety as defined in the Technical Specifications was not reduced.

Did not involve a change in the Technical Specifications as the plant was in compliance with the Technical Specifications throughout the bus outage.

Did not degrade the Security Plan or Quality Assurance Program as these programs are not affected by the 125V DC bus outage.

Did not degrade the Fire Protection System because 125V DC power from the 125V DC B bus is substituted for any 125V DC power normally supplied by the A bus. Continuous firewatches were stationed in any areas of the plant where fire detection is temporarily lost during the swap of the power supply from A bus to B bus.

Performance of the TOP facilitated breaker maintenance on 71BCB-2A which can reasonable be expected to increase the reliability of said breakers and hence the "A" train DC system. Reasonable and prudent measures were taken to maintain vital "A" train load availability while 71BCB-2A was out of service. In addition, a portable DC charger was pre-staged for use in the unlikely event of the loss of the B"B charger.

The safety evaluation determined that the test was not described in the FSAR and concluded that the action did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-245

JAF-SE-92-143, Rev. 0 **UPGRADE THE PERFORMANCE CAPABILITY OF RWCU SUPPLY INBOARD ISOLATION VALVE OPERATOR**

The purpose of this modification was to upgrade the thrust capability of 12MOV-15 from 10,976 pounds of thrust to 13,305 pounds of thrust at the recommended torque switch setting to comply with NRC Generic Letter 89-10. This was accomplished by replacing the existing motor rated at 10 ft-lb with a motor rated at 15 ft-lb on the existing Limitorque (Model SB-00) operator.

This modification to the Reactor Water Clean Up System Valve 12MOV-15 replaced the motor (from 10 ft-lb to 15 ft-lb) for the existing limitorque operator. This replacement motor meets or exceeds the original specification requirements for the operator. This operator is environmentally qualified to the requirements of 10CFR50.49.

The valve operator has been evaluated for the higher thrust and torque loads (reference 12.1) and found to be acceptable.

A stress analysis have been performed by the valve manufacturer for valve/operator combination (Reference 12.2) and found to be acceptable. The existing stress analysis of the piping and pipe supports was performed in accordance with ANSI B31.1 (1967) was evaluated for the slight increase in motor weight (5 lbs) and found to be within design code allowable limits.

The safety evaluation determined that the modification did not described in the FSAR and concluded that the action did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-111

JAF-SE-92-144, Rev. 0 PRE-OPERATIONAL TEST (POT-10AC) FOR TRAIN "B" RHR VALVES FOR MOD M1-92-111

Preoperational test POT-10AC was performed to verify the design function (to open and close during accident conditions) operability of RHR Train B Motor Operated Valves 10MOV-16B, 10MOV-25B, 10MOV-27B and 10MOV-66B which have had their valve control circuits modified by Modification M1-92-111 to add a logic bypass switch and white indicating light. In addition, rerouting of cable 1RHRBBC120 to 10MOV-16B did not change the design function. Also, the addition of logic bypass switches were tested for its function to bypass automatic logics.

POT-10AC will not create an unreviewed safety question. Prior to testing any valve, plant conditions were such that each valve can be stroked open and closed without affecting the safety of the plant. Simulated automatic opening/closing of MOVs via adding jumpers and lifting leads to energize/de-energize relays did not affect safe operation of the plant during testing. Upon completion, valves were returned to appropriate position for the current plant condition or expected plant conditions once testing is complete.

The safety evaluation determined that the test did not described in the FSAR and concluded that the action did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-92-272

JAF-SE-92-145, Rev. 1 INSTALLATION OF EMERGENCY LIGHTING UNITS (PHASE II)

The purpose of this modification was to resolve emergency lighting inadequacies discovered during an NRC Fire Protection/Appendix R inspection and a revision to the Appendix R Analysis (Ref. T, Ref. U). This modification is a QA Category M modification. It was determined during testing that existing emergency lighting for various areas requiring operator access/egress or operator actions associated with safe shutdown equipment in accordance with 10CFR50 Appendix R Section III. J are inadequate. The areas covered within the scope of this modification are the Relay Room, Turbine Building, Reactor Building, Admin. Building, CAD Shack, and Unheated Storage Area No. 2. Future modifications will cover additional areas.

It can be concluded that the modification of the Emergency Lighting did not degrade the design basis or functions of the plant equipment and can be performed based on the following conclusions.

> Modification F1-92-272 has enhanced illumination for various areas requiring operator access/egress and equipment operation.
>
> All fire protection guidelines were followed. New fire hazards were not created. Additional combustibles (cable) were not added.
>
> It met all design and material requirements for a QA Category M installation.
>
> The modification had no impact to the Security Plant. The modification met QA Category M criteria and met the Fire Protection System requirements.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION:

JAF-SE-92-146, Rev. 0 EVALUATION OF A 10% REDUCTION IN CORE SPRAY PUMP SURVEILLANCE FLOW RATE

The purpose of this safety evaluation was to support a request (Reference 1) to evaluate and prepare changes to the Technical Specifications and the SAFER/GESTR-LOCA analysis for new fuel cycles to decrease the core spray flow requirement. The reduction in the CS flow requirements is intended to allow most or all of the IST program pump performance band to be used to provide the potential for increased system availability.

The safety evaluation is of sufficient scope to demonstrate that the Core Spray (CS) system is capable of performing its intended function and that there would be no effect on the Emergency Core Cooling System (ECCS) licensing basis for the following condition:

> The minimum flow requirement for the CS pump(s) is decreased from 4,625 gpm to 4,265 gpm at a head corresponding to a reactor vessel pressure 113 psi above primary containment pressure.

The functions of the CS System are to provide:

- Core cooling in the event of a Loss-of-Coolant Accident (LOCA) via the core spray injection mode of operation.
- Inventory makeup in the event that reactor inventory is lost.

Each function is assessed assuming a CS pump nominal flow of 4,265 gpm at a head corresponding to a reactor vessel pressure 113 psi above primary containment pressure in the core spray injection mode. The assessment of the CS system performance during LOCA also determines the impact of this change on the ECCS licensing basis.

Based on the above evaluation, it was determined that a decrease of 10% in CS flowrate at a system head corresponding to a reactor vessel pressure greater than or equal to 113 psi above primary containment pressure did not constitute a significant hazard as defined in 10CFR50.92.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-271

JAF-SE-92-148, Rev. 0 SPARING IN PLACE 96PS-14 AND REMOVAL OF ANNUNCIATOR 09-7-3-45

The purpose of this modification was to spare in place pressure switch 96PS-14. The switch measured the pressure of the clean steam from the auxiliary boiler and was no longer required because auxiliary clean steam boiler "B" has been removed from the plant, and the "A" boiler, while still installed, is no longer in service (see memorandum JSEM-92-049, Ref. 12.1). The associated annunciator, 09-7-3-45 "CLEAN SEAL STM PRESS LO", was also removed as part of this modification.

The steam seal system did not perform a safety function and has no impact on the safety analysis contained in the FSAR. The proper QA requirements have been met for the preparation of documents, installation and procurement of materials for this modification.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-255 TEST: POT 13G

JAF-SE-92-151, Rev. 1 RCIC SPEED CONTROLLER 125VDC/48VDC ISOLATED POWER SUPPLY INSTALLATION

Preoperational Test POT-13G was performed to ensure that the LAMBDA Model LFS-40-48 125VDC-48VDC Isolated Power Supply installed as part of Modification M1-92-255, performed its intended function of isolating the RCIC EG-M control system from the effects of grounds on the 125VDC system.

Preoperational Test POT-13G, which verified the ability of the LFS-40-48 power supply to isolate the RCIC EG-M speed control system from the affects of grounds on the 125VDC system, did not create an unreviewed safety question. Testing of the power supply with the plant in the condition as outlined in the POT-13G did not violate the limiting conditions of operation for the RCIC system as addressed in the technical specifications. The application of a ground to the 125VDC system to satisfy these testing requirements as outlined in POT-13G did not contribute to the occurrence or consequences of an accident as this is a condition which can occur as stated in the FSAR. Failure of the power supply will not create an unacceptable condition since speed control of the RCIC turbine will be transferred from electrical to mechanical means to ensure continued operation as required. Upon completion of testing, affected valves were returned to appropriate position for the current plant condition or expected plant conditions.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-255 TEST: POT 23L

JAF-SE-92-152, Rev. 1 HPCI SPEED CONTROLLER 125VDC/48VDC ISOLATED POWER SUPPLY INSTALLATION

Preoperational Test POT-23L was performed to ensure that the LAMBDA Model LFS-40-48 125VDC-48VDC Isolated Power Supply installed as part of Modification M1-92-255, performed its intended function of isolating the HPCI EG-M control system from the effects of grounds on the 125VDC system.

Preoperational Test POT-23L, which verified the ability of the LFS-40-48 power supply to isolate the HPCI EG-M speed control system from the affects of grounds on the 125VDC system, did not create an unreviewed safety question. Testing of the power supply with the plant in the condition as outlined in the POT-23L did not violate the limiting conditions of operation for the HPCI system as addressed in the technical specifications. The application of a ground to the 125VDC system to satisfy these testing requirements as outlined in POT-23L did not contribute to the occurrence or consequences of an accident as this is a condition which can occur as stated in the FSAR. Failure of the power supply will not create an unacceptable condition since speed control of the HPCI turbine will be transferred from electrical to mechanical means to ensure continued operation as required. Upon completion of testing, affected valves were returned to appropriate position for the current plant condition or expected plant conditions.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-213

JAF-SE-92-153 MAIN STEAM LINE "A" RADIATION MONITOR CABLE REPLACEMENT

The purpose of this modification was installation of new cables to the Main Steam Line "A" Radiation Monitor. Existing cables 1RPSADX273 and 1RPSADX274 were replaced from monitor 17RM-251A in CAB 09-12 to junction box JB-RPS8 in the Reactor Building. This modification replaced only the "A" channel monitor cables.

AMP TNC Jack Adapters and Plugs were used to join the new cables to existing cables in JB-RPS8. Amphenol type HN Model 82-816 connectors were used to connect the new existing connector. In addition, the new cables were routed in new conduit from JB-RPS8 in the Reactor Building to JB-RPS7 in the relay room. The replacement cables and connectors exceed the original specification requirements for the existing cables. The cable is qualified for Class IE use in accordance with IEEE Stds. 323-1974 and 383-1974.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-277

JAF-SE-92-155 ADDITIONAL HYDROGEN SUPPLY ISOLATION VALVES

The purpose of this modification was to resolve a temporary modification that was previously installed to support the In-core Stress Corrosion Monitoring System Test Program by making it a permanent part of the Hydrogen System Design. This resolution included the review and evaluation of the continued need to make temporary modification #90-160 a permanent installation.

This temporary alteration to the Hydrogen Storage Complex consisted of the installation of two transfer isolation valves, and a connection for attaching a flexible steel bellows hose to provide an outside source for additional Hydrogen. This Modification evaluated and approved the permanent installation of two transfer isolation valves and a hose connection for attaching to an outside temporary Hydrogen supply if needed for start-up or normal plant operation.

There were no changes to the design function of the Hydrogen piping system as a result of adding the globe valves and associated hardware for connecting to an outside Hydrogen source. This modification did not degrade the QA, Fire Protection, Security, or EQ Programs. The FSAR change (Fig. 9.20.1) of this minor modification package does not alter or reduce the margin of safety. The temporary modification did not address the outside source, only the additional valves and hose connection for providing a means of connecting to the outside hydrogen supply. Therefore, the temporary modification is acceptable as a permanent installation and does not constitute an unreviewed safety question pursuant to 10CFR50.59.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-242

JAF-SE-92-156 HPCI VENT AND DRAIN HOSE CONNECTIONS

This modification addressed a deficiency with the draining of the HPCI system. This modification was done to the QA Category II/III portion of the HPCI System. The Equipment/Floor Drain Systems are QA Category II/III systems. No possible nuclear safety hazards are associated with this modification and its installation since no safety related or systems important to plant safety are affected.

The operation of the drain valves on the HPCI system remained unchanged by this modification. The problems of ponding, the spread of contamination and the inability to maintain area cleanliness were minimized. The design and installation of the quick connect couplings, valves and associated piping in accordance with plant standards did not pose a nuclear safety concern.

This modification has been shown not to have an adverse impact on safety related systems and components. In addition, this modification has been shown to meet plant and industry design standards.

The modification to the HPCI system and Equipment/Floor Drainage system did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety. This modification did not create the possibility of an accident or malfunction of a different type than any evaluated previously in the JAF FSAR. There is no reduction in the margin of safety as defined by the basis for any Technical Specifications as a result of this modification.

This modification required changes to the FSAR Fig. 7.4-2, Fig. 9.13-1 and Fig. 9.13-3

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-92-257

JAF-SE-92-157, Rev. 1 CABLE REROUTE IN FIRE AREAS 1A AND VII

The purpose of Modification No. F1-92-257 was to ensure the availability of Motor Control Centers (MCCs) 71MCC-263, 71BMCC-4 and 71BMCC-6 for a fire in Fire Area IA (Zone AD-3), by rerouting the power feeder cables to these MCCs out of Fire Area IA (Zone AD-3).

Also, the purpose of Modification No. F1-92-257 was to ensure the availability of Electric Bay Unit Coolers 67UC-16B1 and 67UC-16B2 for a fire in Fire Area VII, by relocating these loads from MCC 71MCC-261 to MCC 71MCC-262. 71MCC-261 is supplied from common feeder circuit breaker (12606) in 600V Swgr. 71L26 which also supplies 71MCC-263, located in Fire Area VII. Therefore, for a postulated fire in Fire Area VII breaker 12606 in Swgr. 71L26 would trip, de-energizing both 71MC-263 and 71MCC-261. Unit Coolers 67UC-16B1 and 67UC-16B2, required for Electric Bay cooling, would also be lost and alternate shutdown capability may be compromised.

Additionally, two (2) Reactor Protection System (RPS) instrumentation cables were removed from existing conduits to allow space for re-routing of the feeder cables to MCC 71MCC-263. These RPS cables were also rerouted and replaced.

No electrical components (except for cables, conduits, sleeves, trays and junction boxes) were added or deleted as a result of this QA Category I modification. Two (2) existing MCC combination starter units were relocated from 71MCC-261 to 71MCC-262, as part of this modification.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-92-257 TEST: POT 71AC

JAF-SE-92-158 CABLE REROUTE IN FIRE AREAS 1A AND VII

The purpose of this Nuclear Safety Evaluation was to examine the impact of Pre-operational Test Procedure POT-71AC on the plant safety. The test procedure shall be conducted after completion of Modification F1-92-257, to verify the following:

1. Phase sequence and hence, direction of rotation of AC motors connected to 600V AC 71MCC-263 was unchanged by rerouting of incoming feeder cables to 71MCC-263.

2. Functional operation of Main Steam Line Radiation Monitor Loop 17RE-23OD/17RM-251D was unaffected by rerouting of RPS instrumentation cables.

3. Functional operation of Unit Coolers 67UC-16B1 and 67UC-16B2 was unaffected by relocation of starter units from 71MCC-261 to 71MCC-262.

4. Polarity and direction of rotation of DC motors for valves and pumps connected to 125V DC MCC's 71BMCC-4 and 71BMCC-6 remain unchanged by rerouting the incoming feeder cables to 71BMCC-4 and 71BMCC-6.

5. Wiring modification made to supply temporary power before modification and normal power after the modification had not affected the functional operation of the valves, fed from compartments involved in temporary power connection.

This Test Procedure did not conflict with nor degrade the design basis or function of any other plant system or components and can be performed without compromising the plant safety.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-111 TEST: POT 10AD

JAF-SE-92-160 ISOLATION OF "A" TRAIN RHR VALVES

Preoperational Test POT-10AD was performed to verify the design function (to open and close during accident conditions) operability of RHR Train A Motor Operated Valves 10MOV-16A, 10MOV-25A, 10MOV-27A and 10MOV-66A which have had their valve control circuits modified by Modification M1-92-111 to add a logic bypass switch and white indicating light. The addition of logic bypass switches will be tested for their function to bypass automatic logics.

The bypass switches provide a manual means in the control room to bypass the valve logics, allowing the operator to control the operation of these valves in the event of a Reactor Building fire. The white indicating lights illuminate when the bypass switches are in BYPASS.

POT-10AD, which verified the operability of control circuitry for RHR Train A Motor Operated Valves 10MOV-16A, 10MOV-25A, 10MOV-27A and 10MOV-66A which were modified or disturbed by modification M1-92-111, did not create an unreviewed safety question. Prior to testing any valve, plant conditions ere such that each valve could be stroked open and closed without affecting the safety of the plant. Simulated automatic opening/closing of MOVs via adding jumpers and lifting leads to energize/de-energize relays did not affect safe operation of the plant during testing. Upon completion, valves were returned to appropriate position for the current plant condition or expected plant conditions once testing was complete.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: TEST: POT-23M

JAF-SE-92-161 HPCI AND RCIC TORUS SUCTION VALVES REMOTE MANUAL CLOSURE CAPABILITY

Preoperational Test POT-23M verified the operability of the HPCI and RCIC torus suction valves and associated bypass switches installed for remote manual closure capability. The bypass switches and their associated indication light were added to the torus suction valve circuits per modification M1-92-214. The bypass switch allows the torus suction valves to be closed for primary containment isolation when an auto open signal is present. Bypass switch 13A-S25 and indication light 13A-DS66 are associated with 13MOV-39 and 13MOV-41. Bypass switch 23A-S31 and indication light 23A-DS76 are associated with 23MOV-57 and 23MOV-58. The manual control of the torus suction valves was verified first and the automatic actuation logic was verified after the manual control.

The HPCI and RCIC torus suction valves are dual function primary containment isolation valves. They function as a part of our radiation release barrier integrity and they also function as a part of the reactor coolant inventory control system. The ability to close and assure isolation is the most important function the torus suction valves have and the bypass switches assure this function.

This test did not cause an unsafe plant condition because HPCI and RCIC were not required when the reactor was depressurized. The HPCI and RCIC pumps did not run during this test, the torus suction valves were repositioned only. Therefore, the proposed test did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety, did not create the possibility of an accident or malfunction of a different type, did not reduce the margin of safety as defined in the basis for any Technical Specification and did not involve an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-282

JAF-SE-92-162 REWIRE ALARM RELAY FOR ANNUNCIATOR 9-5-1-59

The purpose of M1-92-282 was to rewire alarm relay (KP-1), located in panel 09-43, which provides the input to annunciator 9-5-1-59 (annunciator system on DC power) to alert control room operators as to the status of the annunciator system power supply. Prior to this modification, the annunciator would not alarm if the AC power fuses open and the annunciator system switched to the DC back-up supply. The alarm relay only monitored AC power from the supply distribution panel.

Alarm relay (KP-1) was rewired such that it is located downstream of the AC supply breaker and the AC power fuses. This modification did not require physical relocation of existing equipment and will entail minor wiring changes to accomplish the design purpose.

The Control Room Annunciator System did not have a design or safety bases outlined in the FSAR and is not governed by Technical Specifications. M1-92-282 did not create any new guidelines or affect any pre-existing guidelines in either of these licensing documents. This modification did not increase the probability or consequences of an accident postulated in chapter 14 of the FSAR.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-92-273

JAF-SE-92-163 DEMOLITION OF ENCLOSURES 71INV-3A, 3B AND 71L15, 16

The purpose of this modification was to remove the environmental enclosures and associated air conditioning equipment from the 600V Emergency Power Switchgear Substations (71L-15 and 71L-16) and the LPCI Independent Power Supply Charger/Inverter (71INV-3A and 71INV-3B).

The modification to remove the environmental enclosures and air conditioning system from the LPCI Inverters and Switchgear Substation did not increase the probability of occurrence or consequence of an accident or malfunction of structures, systems, or components important to safety previously evaluated in the FSAR. In addition the modification did not create the possibility of an accident or malfunction of a different type than evaluated in the FSAR. The modification to remove the environmental protection from the LPCI Inverters and Switchgear Substation was done because the protection will no longer be required. The LPCI Inverters were taken out of the EQ Program because an alternate power supply can be used to power the LPCI MOV valves. The Switchgear Substation was environmentally qualified for the postulated casualties and no longer requires the enclosure environment.

The JAF FSAR has been reviewed for applicability to this modification. FSAR sections 7.1.2, 8.10.3, 9.8.3.3, 9.9.3.2, 9.9.3.3, and Figure 12.3-3 and 12.3-5 describe the operation of the air cooled condensing unit (ACCU), air handling unit (AHU) and control panel (PNL). The applicable areas of the sections listed above will be deleted since the air conditioning equipment is being removed by this modification.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-92-269

JAF-SE-92-165 INSTALLATION OF EMERGENCY LIGHTING UNITS

The purpose of this modification was to resolve emergency lighting inadequacies discovered during an NRC Fire Protection/Appendix R inspection and a revision to the Appendix R Analysis. This modification is a QA Category M modification. It was determined during testing that existing emergency lighting for various areas requiring operator access/egress or operator actions associated with safe shutdown equipment in accordance with 10CFR50 Appendix R Section III.J were inadequate. The areas covered within the scope of this modification are the Reactor Building, Emergency Diesel Generator Building, Screenwell House, Relay Room and the East and West Electric Bays.

Exide F-100 and Holophane M-19 Emergency Lighting Battery/Charger packs were connected to existing normal light circuits to provide additional lighting in the required areas. This configuration provides emergency lighting for a minimum of 8 hours continuous operation upon the loss of normal lighting power.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A (Temp Mod)

JAF-SE-92-169 TEMPORARY REPLACEMENT FOR 71T-13

This Nuclear Safety Evaluation addresses the temporary installation of a QA Cat. I 750 KVA transformer in place of the QA Cat I 1000 KVA transformer for 71T-13 in accordance with WACP 10.1.3 and MCM-4. The 1000 KVA transformer normally installed at 71T-13 experienced damage resulting from a three (3) phase ground fault during outage related restoration of the 10500 Bus following maintenance activities. The damaged 1000 KVA transformer was returned to the manufacturer (G.E.) for root cause failure analysis and repair/rewind. An environmentally qualified (EQ) 1000 KVA transformer will be reinstalled at 71T-13 prior to plant startup.

The interim replacement of the original 1000 KVA transformer for 71T-13 with a spare QA Cat I 750 KVA transformer was acceptable to support refueling and outage activities prior to plant startup. Administrative load controls ensure adequate capacity, with a conservative margin, to power the Technical Specification required equipment during this period. The transformer is QA installed location in the 71L 15 load center.

Postulated environmental qualification conditions during this interim period were reviewed and found to be below the threshold for a harsh environment; thus, qualification per 10CFR50.49 is not required or necessary. The Temporary Modification did not involve an unreviewed safety question and is acceptable and the associated Technical Specification required equipment can be declared operable consistent with the plant conditions described in Section III. The 71T-13 transformer was returned to its original condition prior to plant startup.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-92-078 TEST: POT-93N

JAF-SE-92-170 EDG JACKET COOLER LINE REROUTE

Modification No. F1-92-068 rerouted the ESW return line piping from the EDG Jacket Coolers to the C.W. Discharge Tunnel. Subsequent post work testing showed that flow to the EDG coolers was higher than the acceptable limits. The orifices sizes were changed to reduce flow to the EDGs and a new preoperational test must be performed to demonstrate that the new flows are acceptable prior to declaring modification number F1-92-068 complete and the EDGs operational.

The purpose of this safety evaluation is to document the acceptability of performing this new preoperational test in lieu of performing ST-8Q to confirm that sufficient flow to the EDGs coolers is available and is acceptable to make the EDGs operational.

A post work test requirement of modification number F1-92-068 was to perform ST-8Q. System availability does not allow performing ST-8Q prior to the time that the EDGs are required to be operational. This pre-operational test was performed to confirm that the EDGs are receiving proper flows.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-92-294

JAF-SE-92-171 UPGRADE OF SHUTDOWN COMMUNICATION SYSTEM

The purpose of the Nuclear Safety Evaluation (NSE) was to examine the impact of modification F1-92-294 on plant safety. The modification extended the Plant Shutdown Communication System to Control Room Panel 09-3 and Instrument Rack 25-51 and upgraded the existing system to QA Category M in support of Appendix 'R' safe shutdown of the plant. This modification also ensured uninterrupted communication between Control Room and the Shutdown Panels in case of fire in Fire Areas 8 and 9, by the addition of an Uninterruptable Back-up Power Supply and circuit isolation fuses to the Plant Shutdown Communication System.

The Plant Shutdown Communication System is functionally QA Category M. However, as the Communication System is powered from QA Category I Division 'B' Power, the modification is classified QA Category I, the highest safety class of equipment being modified.

Modification F1-92-294 extended the plant Shutdown Communication System to Control Room Panel 09-3 and Instrument Rack 25-51 and upgraded the existing system to QA Category M in support of Appendix 'R' safe shutdown of the Plant. This was necessary as an Alternate Appendix 'R' Reactor Water level gauge was installed in Rack 25-51. This level gauge is to be used in case of fire in Fire Areas 7, 8 and Elevation 326' of Fire Area 9, when there is a possibility of creating erroneous readings in the Reactor Water Level instrument loop indication in the Control Room. Further the Mod. provides for uninterrupted communication between the control room and shutdown panels in case of fire by the addition of an uninterruptable back-up power supply and circuit isolation fuses to the plant shutdown communication system circuitry.

This safety evaluation reviewed the potential impact on plant safety due to the addition of the communication stations in Control Room panel 09-3 and Instrument Rack 25-51.

The Plant Shutdown Communication System has been re-classified as QA Category M. Since the system derives power from a Div. B power source, this modification is classified QA Category I.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-89-091

JAF-SE-92-172 MCC AND BMCC OVERLOAD RESET PUSHBUTTON REPAIRS

This modification resolved WR # 55909, 56897, 64774, 68868, 69296, 69297, 109017. In addition any future problems involving broken or inoperative overload reset buttons shall be resolved in accordance with this modification.

The purpose of this modification was to repair the reset pushbutton assembly for MCC and BMCC.

This modification repaired the overload reset pushbutton assembly in MCC and BMCC door by removing damaged assembly and replace with the plate which covers the hole in the door.

Installation of the plate did not impact or affect function or operability of MCC and BMCC because of the following:

1. The plate function is to cover the hole in the door.

2. The personnel can reset the overload relay individually by pushing the overload relay pin inside of MCC and BMCC cubicle.

This modification did not affect the JAF FSAR or the JAF Technical Specifications because the overload reset pushbutton in MCC and BMCC door is not a subject of discussion in JAF FSAR and Technical Specification. It is not a subject of the Security Plan, Quality Assurance Program or the Fire Protection System requirements.

No unreviewed safety and environmental questions, concerning this modification, exist with installation of the plate which covers the hole in the MCC and BMCC door.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-299

JAF-SE-92-173 DISTRIBUTION PANEL 71DC-A3 FUSE ADDITION

The purpose of this modification was to address 125VDC power circuit selective coordination concerns with feeder circuits from DC Power Panel 71DC-A3 and Control Board 71BCB-2A. The inherent characteristics of circuit protective devices for Panel 71DC-A3 feeder circuits and feeder from Control Board 71BCB-2A to the same panel creates a potential for loss of the entire panel in case of a short circuit fault on either #5 or #8 circuit.

An evaluation of the 125VDC system protective device coordination for DC Panel 71DC-A3 disclosed a coordination problem with circuits #5 and #8. Breakers for Circuits #5 and #8 were not coordinated with their upstream breaker. A fault on either circuit could create a situation where Panel 71DC-A3 feeder breaker in Battery "A" Control Board 71BCB-2A trips before the upstream circuit breaker trip. This would make Panel 71DC-A3 inoperable and create the loss of safety-related circuits fed from this panel.

New fuses have characteristics that support the coordination of cascaded protection of 125VDC system. A new fuse box with four Bussmann Type "KWN-R-20" fuses was installed next to the Panel 71DC-A3 inside Relay Room, floor EL. 286'0". Circuits #5 and #8 were disconnected from panel 71DC-A3 and reconnected thru the new fuses. A new short section of cable will be installed between each fuse attached to the DC power panel with a short piece of flexible conduit. The new installation will meet QA Category I requirements.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-92-329

JAF-SE-92-174 INSTALLATION OF REWOUND (1000 KVA) TRANSFORMER IN 71T-13 MOD. NO. POSITION AND STRUCTURAL MODIFICATION OF 71t-14 (1000 KVA) TRANSFORMER

The purpose of modification no. F1-92-329 was to install a rewound 1000KVA transformer in the 71T-13 position. This required that a 750 KVA transformer, temporarily installed at the 71T-13 position, be removed. The 750 KVA transformer was installed by Temporary Modification No. 92-242 as an interim replacement to provide a safety-related source of power for the 71L15 load center while the 1000 KVA transformer normally installed in the 71T-13 position was being rewound. In addition, this modification documented the as-built bolt/weld modifications made to the 1000 KVA transformer installed in the 71T-14 position as well as documented the weld modifications to the disconnect switch and switchgear enclosures associated with both 71T-13 and 71T-14 performed under Temporary Modification Nos. 92-291 and 92-292.

The installation of the rewound 1000 KVA transformer for 71T-13 was considered acceptable. This was based on the transformer being functionally equivalent to the original 1000 KVA transformer and having been qualified by the transformer vendor who performed the rewind. The lower impedance of the rewound 1000 KVA transformer has been determined to be acceptable since the higher short circuit currents are within the minimum rating s of the protective devices on the transformer secondary and will have no affect on protective device coordination. The seismic installation of the rewound 1000 KVA transformer is satisfactory for the 71T-13 position. The weld modifications made to the 1000 KVA transformer at 71T-14 and the disconnect switch and switchgear associated with both 71T-13 and 71T-14 provide a seismic installation that is satisfactory.

As determined by the analysis and review, the probability of occurrence or consequences of an accident or malfunction of structures, systems, or components important to safety previously addressed in the FSAR have not increased. Therefore, Modification No. F1-92-329 does not involve an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-214 TEST: POT-14D

JAF-SE-92-175 CORE SPRAY OUTBOARD INJECTION VALVES REMOTE MANUAL CLOSURE CAPABILITY

Preoperational Test POT-14D verified the operability of the Core Spray Outboard Injection valves and associated bypass switches installed for remote manual closure capability. The bypass switches and their associated indication light were added to the outboard injection valve circuits per modification M1-92-214. The bypass switch allows the outboard injection valves to be closed for primary containment isolation when an auto open signal is present. Bypass switch 14A-S16A and indication light 14A-DS35A are associated with 14MOV-11A. Bypass switch 14A-DS35A are associated with 14MOV-11A. Bypass switch 14A-S16B and indication light 14A-DS35B are associated with 14MOV-11B. The manual control of the outboard injection valves was verified first and the automatic actuation logic was verified after the manual control.

With the reactor in the cold condition a minimum of two low pressure ECCS subsystems of four must remain operable if work was being performed with the potential for draining the reactor. A minimum of one low pressure ECCS subsystem was required if no work is being performed with the potential for draining the reactor. This test only worked one core spray at a time. The Core Spray pumps were not run during this test, the outboard injection valves were repositioned only and the inboard injection valves had their breaker opened and remained closed at all times during testing for primary containment isolation. This test does not involve an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-331 TEST: POT-73A

JAF-SE-92-177, Rev. 1 SERVICE WATER AND W. DIESEL FIRE PUMP ROOM VENTILATION

The purpose of Modification M1-92-331 was to establish interim compensatory measures to assure proper ventilation for the Safety-Related Pump Rooms until a long-term evaluation and modification can be performed. This was in compliance with a temporary exemption request, which was approved by the NRC, for not modifying the Safety-Related Pump Room Ventilation System for Appendix R.

The ventilation system was restored to it's originally designed functional configuration.

The Modification reinstituted the original ventilation design which has been evaluated to be still valid. The removal of the fire protection interlocks did not increase the possibility of fire or other accident from prematurely closing the dampers and causing loss of ventilation.

This modification did not degrade the security plan because the security grates on the ventilation openings were not altered and there are no security devices on the two fire doors. The Quality Assurance program was not degraded because everything will be done in accordance with the applicable Quality Assurance criteria. The fire protection system was not degraded because the actions described provided an equivalent level of protection. Early fire detection was assured by thermal and smoke detectors, fire watches, and high area temperature detectors which will alarm in the Control Room. The combustible-free fire areas helped prevent any fire from starting, spreading, or affecting the safety-related areas.

The Pre-Operational Test POT-73A will ensured that the Modifications were performed correctly and that the ventilation system works correctly. The pre-operational test did not affect the safety of the plant because the room temperature was monitored continuously. The test was to be terminated and room ventilation restored within ten minutes if the temperature exceeds 85°F or any pup, except the jockey pump, in the tested room started.

The Pre-Operational Test POT-73A tested the replaced thermal detectors (WR 112847) 76TAD-1A to 1K in accordance with the manufacturers recommended post work instructions to ensure that panel 76CP-SRP properly received the alarms as designed. The replacement of the detectors did not degrade the fire protection system since the detection system was being returned to its original condition as indicated on the design drawings. The new thermal detectors are the proper detectors, as recommended by the manufacturer, for the ZA-30 module in Fire panel 76CP-SRP.

The safety evaluation concluded that the modification and pre-op test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: F1-92-328

JAF-SE-92-179 INSTALLATION OF EMERGENCY LIGHTING (PHASE IV)

The purpose of this modification was to resolve emergency lighting inadequacies discovered during an NRC Fire Protection/Appendix R inspection and a revision to the Appendix R Analysis. This modification is a QA Category M modification. It was determined during testing that existing emergency lighting for various areas requiring operator access/egress or operator actions associated with safe shutdown equipment in accordance with 10CFR50 Appendix R Section III. J are inadequate. The areas covered within the scope of this modification were the Reactor Building, Emergency Diesel Generator Building, Relay Room and the Battery B Charger Room.

It can be concluded that the modification of the Emergency Lighting did not degrade the design basis or functions of the plant equipment.

The following sections of the FSAR required update:

9.16.1
9.16.2
9.16.3
9.16.4
9.8.3.7

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: **TEST: ST-390**

JAF-SE-92-180 **RECIRCULATION PUMP SPEED LIMITER BYPASS TO ALLOW RECIRCULATION PUMP SPEED INCREASE TO 50% DURING ST-390**

This Nuclear Safety Evaluation addressed the use of the Reactor Water Recirculation (RWR) pumps at levels above minimum-speed during the performance of ST-390 - Reactor Pressure Vessel System Leakage Test with Reactor Coolant Temperature Below 212°F (ISI)*. The RWR pumps may need to be run at higher than the normal, minimum flow rates to develop additional fluid heat input to maintain RPV beltline temperatures above the minimum temperature specified in the JAF Technical Specifications. The potential necessity for providing additional heat input to the vessel water is due to the lack of available decay heat in the reactor core due to the extended JAFNPP refueling outage.

The use of the RWR pumps at up to 50% speed in ST-390 to provide sufficient heat input to maintain the necessary temperature levels was acceptable. Bypassing the #1 RWR pump speed limiter to allow increasing pump speed from minimum to 50% did not result in RWR or jet pump cavitation problems, and did not affect plant safety since the plant conditions for which the limiters were provided did not exist during the test. The use of the pumps for this purpose is specifically identified in the FSAR, and operation of the pumps was in accordance with the normal Plant Operating Procedures. There was no adverse effect on core thermal parameters, RPV structural integrity, RWR and jet pump hydraulic and structural limits will not be exceeded, and flow induced vibration of vessel internals did not exceed analyzed limits. Available Operating Industry Experience concerning RWR pump shaft cracking was reviewed; operating the pumps in this manner during ST-390 will not increase the potential for pump shaft cracking.

The use of the RWR pumps at speeds up to 50% rated, and the bypassing of the #1 speed limiter in ST-390 acceptable, and did not involve an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION:

JAF-SE-92-181, Rev. 0 125V-DC COORDINATION AND SEPARATION CONCERNS

The purpose of this safety evaluation was to address coordination issues identified by the US NRC's Safety System Functional Inspection (SSFI) Report NRCI-89-80. The NYPA produced 125V DC Electrical Distribution System Coordination Study Report No. JAF-RPT-ELEC-00164 identifies those issues raised by the NRC Report and makes recommendations to resolve these coordination issues. This NSE provides the details of each coordination issue raised by the NYPA report, the modification to resolve this issue, and whether these physical hardware changes were required to keep the plant in a safe operating condition.

As per the analysis outlined in this safety evaluation, all instances of miscoordination outlined in the NRC's SSFI Report have been resolved either by hardware replacement, short-circuit current reduction, or evaluation of impact to plant safety analysis.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION:

JAF-SE-92-182, Rev. 0 EVALUATION OF PARTS LOST IN REACTOR CAVITY DURING RELOAD 10 REFUELING OUTAGE

The purpose of this evaluation was to examine the potential effects on nuclear safety of parts lost in the reactor cavity which were unaccounted for at the end of the Reload 10 refueling outage and must therefore be assumed to be in the reactor pressure vessel (RPV).

The parts did not adversely effect reactor water chemistry or system performance, therefore the occurrence of an accident previously analyzed was not increased. There was no impact on Control Rod Drive system performance as a result of loss of these parts, therefore the consequences of an accident remained the same as previously analyzed.

Events resulting in reactor coolant flow decrease were previously evaluated and the dimensions of the lost parts are such that no bundle flow blockage large enough to result in significantly adverse fuel thermal hydraulic performance (onset of transition boiling, fuel centerline melt, fuel cladding melt) was expected.

The dimensions of the lost parts were such that no bundle flow blockage large enough to result in significantly adverse fuel thermal hydraulic performance was expected. If such degradation was assumed to occur, the safety of the plant was maintained by the limiting conditions of operation and limiting safety system settings of the plant. Among the purposes of reactor coolant chemistry surveillance requirements is the assurance of fuel cladding defect detection. Operating limits are specified to ensure appropriate action is taken in response to indications of fuel cladding defects. In the event fuel cladding failures occur on a time scale which precludes detection and correction as a result of normal reactor water chemistry monitoring programs, offgas system and main steam line isolation on high radiation will limit the release of radioactive nuclides to acceptable levels.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-246

JAF-SE-92-183 INCREASE SETPOINT OF 01-107FS-108H FLOW SWITCH

The purpose of this modification was to document the basis for a setpoint increase to 50 SCFM of the flow switch 01-107FS-108H (Recombiner High Flow).

The flow switch 01-107FS-108H trips the "Off Gas Recombiner Trouble" alarm 09-3-1-28 on the annunciator panel 09-3-1 in the Control Room. The setpoint of this switch was changed to 50 SCFM.

The changes in this modification only affect non-safety instrumentation used for alarm signal purposes rather than control. In addition this modification did not increase the possibility of an accident or malfunction of a type other than any evaluated previously in the FSAR. This modification did not require a change to the Technical Specifications. No unreviewed safety questions pursuant to 10CFR 50.59 has been presented by this modification.

MODIFICATION: M1-90-242

JAF-SE-92-184, Rev. 1   CAD SOV REPLACEMENT

This modification consisted of the replacement of solenoid pilot valves on the CAD system air operated valves. The existing SOV's were manufactured by ASCO and have reached the end of their service life. The replacement of the existing SOV's was accomplished in two steps. The first step was to replace the existing valves with valves that had been evaluated to have a minimum operating temperature of 0°F This was acceptable because the SOV's were not anticipated to operate at 5°F or lower prior to the completion of Step 2 on or before November 30, 1992. Before November 30, 1992 a change to this modification package was issued providing instructions on the means for accepting the stock SOV's until permanent replacements arrived or modifying the existing installation to provide an acceptable environment for reliable operation of the stock SOV's.

The final step is to replace the valves installed in step 1 with valves that have a minimum operating range of -40°F.

Section 5.2.3.8 of the FSAR had been reviewed and it was determined that this modification did change the purpose and function described.

The installation of the replacement solenoids have shown not to have an adverse affect on safety related systems and components. In addition, this modification has been shown to meet the plant and industry design standards.

Isolating the nitrogen supply to 27SOV-127A, B during normal operation had been evaluated. Isolating the SOV ensured that if the SOV failed, nitrogen inventory would not be lost. During containment inerting nitrogen would be restored to the SOV. This allows the low level and low temperature trips to operate as designed. Inspection for SOV leakage during inerting assures prompt isolation of the SOV if excess leakage occurs.

a. The margins of safety during normal operation and transient conditions anticipated during the life of the station were not reduced.

b. The structures, systems and components provided for the prevention of accidents and the mitigation of the consequences of accidents continue to remain adequate.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-339

JAF-SE-92-186 **REMOVAL OF RESTRICTING ORIFICES 10RO-105A/B, 14RO-102A/B AND 27RO-113, 117**

This modification evaluated the acceptability of RHR-SW, Core Spray and the CAD system operation <u>without</u> restricting orifices 10RO-105A&B, 14RO-102A&B and 27RO-113, 117 installed. These orifices were recently verified as <u>not</u> installed.

The JAF FSAR, Plant drawings and other Plant records describe the above mentioned restricting orifices being installed in their respective systems. Investigation of available documents indicated that these orifices were removed in an attempt to gain additional flow during Surveillance Testing or removed during original construction.

The functions of the restricting orifices was to limit flow rate to maintain system flow requirements of individual systems. It has been verified by surveillance tests and operation of individual systems that each system still maintained the flow requirements without the presence of the restricting orifices. Performance of individual systems have been proven satisfactory without the presence of RO's by various surveillance tests.

FSAR Fig 7.4-5 and 7.4-7 were updated as a result of this modification.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-334

JAF-SE-92-187 DISABLE VALVES 10MOV-36B, 70A, 70B, 15MOV-101, 102, 103

This modification disabled valves 10MOV-36B, 10MOV-70A, 10MOV-70B, 15MOV-101, 15MOV-102, and 15MOV-103 by revising plant procedures to maintain their associated breakers in the open position. This was required due to possible fire induced failures which could inadvertently open these valves therefore adversely affecting the safe shutdown capability.

The RHR valves are only opened for the steam condensing mode of the RHR system. This mode is a non-safety function and is not utilized to mitigate any Design Basis Accident.

The valve 15MOV-101, ESW A and B cross connect, is not part of the safety related design basis of the ESW Systems since the safety related loads supplied by each independent loop are 100 percent redundant. This capability is provided to increase flexibility at operator discretion. Valves 15MOV-102 and 15MOV-103 provide ESW flow to the Division A and B drywell coolers. These are large non-safety related cooling loads on the ESW system and may starve safety related loads if flow is initiated.

The affected valves are presently normally closed motor operated valves which require operator actions to open. This modification did not revise the valve position, but opened their respective breakers for each valve. This will prevent spurious valve operation which has been postulated during an Appendix R Fire. Valve operability could be restored by closing the breaker for the desired valve.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-336

JAF-SE-92-188 DEMINERALIZED WATER HOSE CONNECTION

The purpose of this modification was to install a short branch connection with a quick disconnect hose fitting. This will provide plant personnel a source of Demineralized Water for Neutron Shielding and RWCU Decon.

This modification approved the permanent installation of a short branch connection to the Water Treatment System for providing Demineralized Water at the 272'-" El. of the Reactor Building. This modification, which is non-safety related and non-seismic, did not alter the design or safety function of the Water Treatment System. The modification will not degrade the QA Fire Protection, Security, or EQ Programs.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION:

JAF-SE-92-189 FREEZE SEAL TO REPLACE RECIRCULATION ISOLATION VALVE 02-2RWR-51A

The purpose of this evaluation was to review the use of a nitrogen freeze seal on the nuclear boiler vessel instrument line 3/4"-I-1504-104 stainless steel piping for the isolation and replacement of valve 02-2RWR-51A, RWR jet pump instrument nozzle N-8A penetration reactor recirculation system isolation valve. This valve was unisolable from the reactor vessel.

The work; (a) was performed using the same or more stringent construction codes and requirements than those stated in the FSAR, (b) did not change any inputs to a FSAR transient or accident analysis.

The application did not introduce any new failure mode or component/system interaction and a leak due to freeze seal failure is enveloped by the FSAR LOCA analyses.

The proposed application did not degrade the Security Plan, Quality Assurance Program or Fire Protection System because this application did not impact the Security Plan or the Fire Protection System and was performed in accordance with the Quality Assurance Program.

The proposed application did not affect the environmental impact of the plant or involve an unreviewed environmental question because this application did not result in any discharge of effluents in excess of those addressed by current Radiological Effluent Technical Specifications.

The safety evaluation concluded that the action did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION:

JAF-SE-92-190 QA CATEGORY CLASSIFICATION CHANGE - MCM-6A, PARAGRAPH 4.1.1.4 OF ATTACHMENT 4.8

This safety evaluation was prepared to support a procedural change to MCM-6A, Component Classification and System Safety Function Control - JAF, which affects the classification criteria for the pressure boundary for QA Category M flow paths. A revision of the criteria, for the Fire Protection System, has been made to allow portions of the Fire Protection Systems which do not supply safety-related fire suppression systems, to be excluded from the QA Category M Classification even though they are connected to the system via normally open valves.

The change to the Classification Criteria for Category M flow paths contained in section 4.1.1.4 of MCM-6A allows portions of the Fire Protection Water Supply System serving non-safety related areas (i.e. outside the power block) to be classified less than QA Category M. This change did not adversely affect previous analyses in the FSAR, nor did it affect the Technical Specification or its bases. The change did not degrade the Security Plan, Quality Assurance Program or the Fire Protection System for safety-related areas. Portions of the fire protection water supply system serving non-Category M areas will be isolable from the remainder of the system by two QA Category M manual isolation valves.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION:

JAF-SE-92-209 EVALUATION OF TOP-144, RPV INSTRUMENT LINE REWORK

Temporary Operating Procedure TOP-144 provided guidance for removing various reactor pressure vessel (RPV) water level instrumentation from service during rework of reactor pressure vessel instrument tubing 02-3-3/8"-I-43 and 02-3-3/8"-I-49.

Temporary Operating Procedure TOP-144 had been developed to control the removal from service of instruments supplied by the affected instrument tubing and directed use of alternate reactor water level indication. There were no unreviewed safety questions associated with the performance of TOP-144 and adequate precautions existed to prevent inadvertent initiation of safety systems. Adequate reactor vessel water level indication was provided by redundant wide and narrow range instrumentation that will remain in service during the tubing rework.

The safety evaluation concluded that the action did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: **TEST: STP-76AI**

JAF-SE-92-213 **SPECIAL TEST FOR APPENDIX R EMERGENCY LIGHTS**

Special test STP-76AI was performed to ensure the effectiveness of emergency lighting units used to illuminate 10CFR50 Appendix R equipment and access/egress routes at JAF. All Appendix R emergency lighting units are QA Category M and were functionally tested to determine if the equipment associated with the emergency light unit or access/egress paths were adequately illuminated. Normal area lighting was de-energized and non-Appendix R emergency lighting units were placed in standby during the test. Appendix R Emergency light unit heads were permanently fixed and marked during the test, in their required positions.

Special Test STP-76AI, which verified the adequacy of the illumination levels provided by the emergency lighting units used to illuminate 10CFR50 Appendix R equipment and access/egress routes at JAF, did not create an unreviewed safety question. Testing of the lighting with the plant shutdown in the cold condition as outlined in STP-76AI did not violate the Technical Specification. The test of the Appendix R emergency lighting did not contribute to the occurrence or consequences of an accident as stated in the FSAR. Upon completion of STP-76AI, all normal lighting was restored.

The safety evaluation determined that the test was not described in the FSAR and concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION:

JAF-SE-92-216 SEISMIC INSTALLATION OF 71T-13 AND 71T-14 TRANSFORMERS

The purpose of this nuclear safety evaluation was to evaluate the adequacy of the seismic qualification, including anchorage, of the 71T13 and 71T14 transformers and the adjacent breaker and disconnect switch sections of the 71L15 and 71L16 substations, all located at El. 300′ of the Reactor Building. The anchorage of the transformers and the adjacent breaker and disconnect switch sections were modified by Temporary Modification Nos. 92-291, and 92-292.

During a walkdown performed for transformer 71T13, it was found that the transformer was not anchored to the concrete slab. The same condition existed for the 71T14 transformer. Due to absence of information on original anchorage requirements and based on subsequent transformer designs, General Electric (the original equipment supplier) recommended that the transformers be anchored to the concrete floor. Anchorage was designed and the modification was installed to seismically qualify the complete 71L15 and 71L16 substations to their original design basis. Therefore, these components, with the anchorage modification, are seismically qualified for OBE and DBE.

The safety evaluation determined that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-351

JAF-SE-92-217 DISABLING OF 70TCV-120A&B

The purpose of this modification was to disable the actuators for temperature control valves 70-TCV-120A, 120B, 121A and 121B. These valves are safety related QA Cat. I and serve to modulate the chilled water or service water for the Control and Relay Room Air Handling Units, 70-AHU-3A, 3B, 12A and 12B. The valves will remain in the fail open position, allowing maximum cooling water flow, as an interim solution while the subject valve actuators are either repaired or replaced. The manual bypass valves, 70WAC-15A, 15B and 22A and 22B, will also be closed to ensure full flow to the air handling units.

Interim temperature control will be provided by manual means with control of the in-duct heaters and cooling water flow to the air handling units as required. Based on the review of the FSAR, Technical Specifications and proposed interim temperature control methods, it is concluded that the modification described above will not present an unreviewed safety question, will not degrade the safety function of the Control and Relay Room Ventilation system, and will provide adequate temperature control without compromising plant safety. It is further noted that this modification does not involve an unreviewed environmental question or impact the environmental aspects of the plant.

The safety evaluation determined that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A

JAF-SE-92-218, Rev. 0 REVISION TO MAIN STEAM OPERATING PROCEDURE (OP-1)

The purpose of this safety evaluation was to demonstrate that maintaining reactor vessel water level utilizing the main steam line drains is an acceptable alternate method for shutdown level control. This method is desirable in order to provide an alternate flow path to remove water entering the vessel from either the Control Rod Drive Mechanism cooling flow or the Feedwater System.

In conclusion, this procedural change to Main Steam Operating Procedure (OP-1) that allows maintaining reactor vessel water level utilizing the Main Steam line drains did not conflict with the design basi of the Main Steam system as stated in the FSAR Section 4.11. . so, it was determined after a review of FSAR sections 14 and .5 that this change did not constitute an unreviewed safety question or result in changes to the JAFNPP Technical Specifications. Finally, this procedure change to OP-1 did not impact on any safety related or environmentally qualified structures, systems or components because the procedure change has been evaluated for postulated equipment failures and no new scenario exists.

MODIFICATION: M1-92-383

JAF-SE-92-220 ENCAPSULIZATION OF 12RWC-46 EQUALIZATION LINE

This minor modification was required to repair a 1/2" diameter schedule 80 stainless steel bonnet equalization line on a 6" diameter carbon steel reactor water cleanup (RWCU) manual isolation valve 12RWC-46 located inside the drywell. This 1/2" pipe had experienced a through-wall circumferential crack at the toe of the weld attaching the line to the bonnet portion of the 6" RWCU Isolation Valve. This modification repair will be removed during the next refuel outage.

The repair of 1/2" diameter schedule 80 stainless steel line by encapsulating it with a larger diameter pipe restored pressure boundary and reinforced the piping. The design, installation, examination and testing met the requirements of ASME Section XI as described in the repair plan document. Non Destructive Examination was performed on all the ½" equalization piping fittings and welds to confirm the integrity of these items. No further unacceptable defects existed in these items. This repair will be removed at the next Refuel Outage.

The following section of the FSAR required update: 16.5.5.1.5.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-405

JAF-SE-92-221 EVALUATION OF COMMON SUPPORTS FOR REDUNDANT CONDUITS AND CABLE TRAYS

The purpose of this nuclear safety evaluation was to determine that there were no unreviewed safety questions and evaluated the basis for and significance of the separate support criteria contained in Stone & Webster's design criteria "Separation Criteria for Safeguard Electrical Circuits--James A. FitzPatrick Nuclear Power Plant, Power Authority of the State of New York," Rev. 3, dated, June 16, 1973.

Based upon a review of general NRC design requirements, plant-specific requirements, and single failure criterion it was concluded that the criteria for separate supports for redundant conduit and cable trays was a conservative design preference, rather than a requirement based on a safety need. Therefore, although the conditions found differed from the original Stone & Webster design criteria, no corrective actions were deemed necessary. Passive failure of Category I supports, including common supports, did not need to be considered in the Design Basis Event.

The use of common supports for redundant conduits and cable trays did not involve an unreviewed safety question because all Category I conduits and cable trays are seismically designed to resist seismic loads and therefore the requirement for independent supports was not applicable. This did not increase the probability of occurrence or consequence of any analyzed or unanalyzed accident or malfunction.

The safety evaluation concluded that the safety evaluation did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A

JAF-SE-92-223 REVISION TO AOP-28

Abnormal Operating Procedure (AOP) "Operation During Plant Fires" AOP-28, revision 3, provides guidance to operations shift personnel to mitigate system and equipment failures which could result from fire in any potential fire area. The goal of the procedure is to provide the operator with the information required to safely shutdown the plant to cold shutdown coincident with a fire. The procedure is a safe shutdown aid in predicting and coping with the "worst case" effects of a fire in each of 23 specific fire areas. The procedure is applicable during all plant operating modes, except when the reactor is shutdown in the cold condition.

Each of the steps in the procedure have been evaluated and an individual basis established for each procedure step. These basis have been reviewed and have been determined to be reasonable and prudent considering the potential effects of the fire damage on the ability to achieve Safe Shutdown.

AOP-28 demonstrates, using the very conservative assumptions implied by the acceptance criteria in Appendix R to 10CFR50, that FitzPatrick can be safely shutdown. The operator actions in AOP-28 initiate and maintain operable at least one train of Safe Shutdown equipment during a fire in any fire area of the plant. The steps prescribed by AOP-28 are consistent with FitzPatrick's EOPs.

The procedure did not call for the operation of equipment in ways outside of their intended design and consequently did not increase the likelihood of malfunction.

AOP-28 did not require the use of equipment or systems outside of their intended function. No new unanalyzed reactor operating modes were prescribed by the procedure. Any scenario which results in reactor water level below the top of active fuel is covered by an NRC exemption or is bounded by other accident analyses.

The safety evaluation concluded that the revision to AOP-28 did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-393

JAF-SE-92-231 OFF GAS HYDROGEN ANALYZER DRIP LEG INSTALLATION

The purpose of this modification was to resolve a temporary modification (#91-187) that was previously installed to provide a drip leg to prevent condensate from entering the Off-Gas Hydrogen Analyzer. This modification included the review and evaluation of the continued need to make temporary modification #91-187 a permanent part of the Off-Gas-Hydrogen System design.

This modification evaluated and approved the permanent installation of a drip leg between valves 01-107OFG-94 and 01-107OFG-220. The drip leg provides a means to verify the Off-Gas Dryers are removing moisture.

There were no changes to the design function of the system as a result of this drip leg installation. The drip leg installation and support were visually inspected for conformance to specifications in accordance with approved JAFNPP plant procedures. Therefore, this modification is acceptable. This did not constitute an Unreviewed Safety Question pursuant to 10CFR50.59.

MODIFICATION: 92-305 Temp Mod

JAF-SE-92-234 INSTALLATION OF TEMPORARY PIPING FROM AUXILIARY FILL LINE OF 10 TON CO_2 TANK FOR SUPPLYING VAPORIZER 76E-7

This evaluation assessed the safety significance of installing a temporary pipe from the normal supply line of the CO_2 vaporizer to the auxiliary fill line of the 10 ton CO_2 storage tank. This installation required a 10CFR 50.59 Nuclear Safety Evaluation as a result of operation of the system other than as described in the FSAR. This safety evaluation was written as a basis for Temporary Modification 92-305 as well as a future revision to OP-11C. Both the temporary modification and the procedure revision accomplish the same functions, the difference being that the procedure revision might not have been accomplished prior to startup so the temporary modification was used for startup.

Implementation of the temporary modification and the subsequent OP-11C revision involved running a pipe from the normal fill line of the 10-ton CO_2 while bypassing the tank float valve. This float valve is defective and blocking the normal tank fill line. Although this resulted in the operation of the system differently than that discussed in the FSAR, it was acceptable based upon administrative controls to be implemented. The temporary piping was consistent with the quality assurance requirements of the existing vaporizer fill line, and there was no seismic concern regarding this additional section of piping. This temporary modification and subsequent procedure revision was acceptable for implementation. This temporary modification and subsequent procedure revision did not represent an unreviewed safety question.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A

JAF-SE-92-236 EVALUATION OF APPARENT ELECTRICAL SEPARATION ANNOMALIES

This safety evaluation addressed potential operability issues associated with eighty one (81) apparent deviations from the electrical separation criteria given in the JAFNPP FSAR. Each of the apparent deviations involved a configuration where field walkdowns indicated less than one (1) foot spacial separation exists between cable, conduit, or cable trays of different color code designations.

Of the eight-one (81) apparent anomalies addressed in this NSE, twenty two (22) were found not to represent violations of the JAF design criteria. The remaining fifty-nine (59) apparent anomalies were segregated into three categories:

A) enclosed-to-enclosed configurations
B) enclosed-to-open configurations
C) open-to-open configurations

Each of the fifty nine apparent anomalies was evaluated for single failure potential either by detailed functional assessment (based on the specific affected cable) or by comparison of the field configurations to previous industry tests. IN comparing JAF configurations to industry test results it has been determined:

1) Industry test involved cable qualified to IEEE-383 and JAF cable (except lighting cable) are either qualified to IEEE-383 or have been evaluated to be equivalent.

2) The involved cables services at JAF are bounded by the services tested.

3) The separation distances for each of the fifty nine apparent anomalies is bounded by the test conditions.

Extensive industry cable separation testing has been performed over the past decade.

In summary, the fifty-nine apparent anomalies do not represent potential "points" of signal failure vulnerability and operation of the facility with those apparent anomalies.

The safety evaluation concluded that the evaluation did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A

JAF-SE-92-238 ALTERNATE ELECTRICAL SEPARATION CRITERIA

This Safety Evaluation addressed utilizing horizontal and vertical cable separation distances of 1 foot-1 foot, respectively, and 1 inch-1 inch between conduits, respectively, as alternate separation criteria on an interim basis to assess cable separation anomalies. The alternate separation criteria is an interim exception to the criteria stated in the JAF FSAR and is based on cable separation testing conducted for other plants. The apparent deviations addressed in this NSE are evaluated using the guidance contained in NRC Generic Letter No. 91-18.

The industry tests simulate and bound the actual plant configurations, the cables tested were equivalent to those used in JAF, the cable service (voltage levels) tested bound those being assessed in the walkdowns and the conservative nature of the original raceway design has no unusual characteristics from that tested. Therefore, the reduced separation distances may be utilized since they were based on testing of typical cable installations at JAF. The reductions in separation distances were based on reviews of the test data which indicate that reduced separation distances would still result in adequate separation and a conservative design.

The alternate separation criteria of 1-foot horizontal and 1-foot vertical separation and 1 inch-1 inch for conduits will be utilized on an interim basis to assess cable separation anomalies discovered at JAF during recent plant walkdowns. Based on review of the walkdown documentation (deviation reports) and the cable types, sizes and configurations at JAF, industry testing at lower separation distances is applicable and is used as the basis of the JAF alternate criteria. Use of this separation criteria will maintain adequate separation between redundant safety function circuits at JAF, as demonstrated by test, and therefore does not result in an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: **TEST: STP-76AJ**

JAF-SE-92-240 **SPECIAL TEST FOR ADDITIONAL APPENDIX R EMERGENCY LIGHTS**

Special test STP-76AJ was preformed to ensure the effectiveness of emergency lighting units used to illuminate 10CFR50 Appendix R equipment and access/egress routes at JAF. All Appendix R emergency lighting units are QA Category M and were functionally tested to determine if the equipment associated with the emergency light unit or access/egress paths are adequately illuminated. Normal area lighting was de-energized and non-Appendix R emergency lighting units were placed in standby during the test. Appendix R Emergency light unit heads were permanently fixed and marked during the test in their required positions.

Testing of the lighting with the plant shutdown in the cold condition as outlined in STP-76AJ did not violate the Technical Specification. The test of the Appendix R emergency lighting did not contribute to the occurrence or consequences of an accident as stated in the FSAR. Upon completion of STP-76AJ, all normal lighting was restored.

The safety evaluation determined that the test was not described in the FSAR and concluded that the test did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: N/A

JAF-SE-92-241 ACCEPTABILITY OF SCREENWELL VENTILATION FANS 73FN-2A, 2B TO MEET DESIGN BASIS FLOW REQUIREMENTS

The installed design configuration of the Screenwell Building ventilation fans 73FN-2A, 2B is such that highly accurate measurements of fan flow are not feasible (flows are turbulent due to lack of suction/discharge ducts). Furthermore, the design and operation of the Screenwell Building Ventilation System is different than that discussed in the FSAR.

The purpose of this Safety Evaluation was to demonstrate that the flow data which is available (original balancing data and recent flow data) has sufficient conservative margin above the safety design basis requirement considering inaccuracies in the flow measurement.

Flow measurements inaccuracies for fans 73FN 2A, 2B and system design and operational differences from the system description in the FSAR.

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. This conclusion is based on the Screenwell ventilation system conforming to the original system design criteria including requirements, and the measured flow data having sufficient margin to account for measurement inaccuracies.

FSAR Section 9.9.3.7 needs to be clarified with respect to the operation of the Screenwell Building Ventilation System.

There are no Technical Specification requirements listed, and because the ventilation system design was not changed.

This surveillance test did not affect the Security Plan or Appendix R requirements. The Quality Assurance Program is not degraded because the new surveillance tests will be added to that plan when completed.

The environmental parameters will not change as a result of this surveillance test.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-405

JAF-SE-92-242 CONTROL ROOM HVAC SUCTION REALIGNMENT

The purpose of this modification was to change the normal alignment of the Control Room Heating ventilation and Air Conditioning (HVAC) emergency filtered air intake from the primary intake line to the secondary intake line.

This modification was a direct result of CO_2 entering the Control Room air space during the relay room CO_2 discharge test. The incident prompted the exiting of personnel without self-contained breathing apparatus, from the Control Room.

In addition to the change in the normal alignment of the Control Room HVAC emergency air intake, this modification changed the procedures for the initiation of CO_2 into the relay room. The revised procedures provide for isolation of the control room prior to actuation of relay room CO_2 deluge.

This modification did not require any physical change to the plant. This change will be a document change only. The system flow diagram 11825-FB-45A (System 70) and operating procedures required revisions.

The components are part of the Control Room ventilation system which is safety related. The realignment of the Control Room HVAC emergency air intake did not modify the limiting conditions for operation or the surveillance requirements. This modification is classified as QA Category I.

No changes to the Technical Specification were required as a result of this modification. This modification required an update to the FSAR, Section 9.9.3.11.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION: M1-92-313 Temp. Mod

JAF-SE-92-244 TEMPORARY MODIFICATION TO REPLACE CHECK VALVE 87AHB-287B WITH AN AIR OPERATED VALVE

The purpose of the safety evaluation was to verify the acceptability of a modification of the plant nitrogen steam vaporizer system. The originally supplied check valve on the discharge of the electrap (87TK-6), condensate return to auxiliary boiler piping, had been replaced with an air operated globe valve. This modification was not previously evaluated or documented. The valve air operator is controlled by the electrap level switch 87LS-131 to open the normally closed valve on high condensate level. The steam vaporizer and auxiliary boiler system are quality assurance category II/III.

The steam vaporizer electrap discharge check valve replacement with an air operated valve was acceptable for the nonsafety related containment purge system. The air operated valve performs the function of the check valve for blowdown of the electrap and also prevents the continuous pressurization of the condensate return line. The design is in compliance with the applicable piping and installation specifications. There are no conditions created which are adverse to the safe operation of the James A. FitzPatrick Nuclear Power Plant.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION:

JAF-SE-92-247 JUMPER EVALUATION FOR SERVICE PLATFORM HOIST LOAD ROD BLOCK

The purpose of this safety evaluation was to evaluate the service platform hoist load rod block interlock to determine if it can be administratively controlled by a jumper when the platform is not installed in the refueling cavity.

Interim use of a jumper to provide a rod block bypass for an idle service platform is substantially the same risk as use of the bypass plug in the original design. This parity can be achieved by administrative controls to ensure the rod block is reestablished during situations in which it is required.

The safety evaluation concluded that the modification did not involve a change in the Technical Specification or an unreviewed safety question as defined in 10CFR50.59.

MODIFICATION:

JAF-SE-92-249 RADIOACTIVE MATERIAL STORAGE ON-SITE (SEA/LAND CONTAINER)

This nuclear safety evaluation was performed to document whether or not the temporary storage of radioactively contaminated structural materials within two steel containers on-site constituted an unreviewed safety question.

The structural materials are re-useable and are not low level radioactive waste.

The radioactive material stored within steel containers is located within the protected area. Radiation exposure to the public and workers is being controlled by surveillance, labeling and by situating the material within the protected area of the site. The radiation level at the boundary of the steel container is limited to 0.5 mR/hr or less.

The items stored are re-useable. The items will remain in the locked, steel containers until re-use is necessary or until another storage area is designated. Until that time the items will be controlled as described in this nuclear safety evaluation.

Based on the above analysis, the storage of contaminated structural materials within the steel container on-site does not constitute an unreviewed safety question as defined in 10CFR50.59.