POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

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May 6, 1982
SERIAL: JAFP 82-0495

United States Nuclear Regulatory Commission
Region I 631 Park Avenue
King of Prussia, Pennsylvania 19406

Attention: Ronald C. Haynes
Director

Subject: Docket 50-333- Annual Summary of JAFNPP Plant Modifications, Changes, and Experiments for 1981

Enclosures: (1) Summary of Major Plant Modifications Implemented at JAFNPP During 1981 and the 1981-1982 Refueling Outage

(2) Summary of Tests and Minor Plant Modifications Implemented at JAFNPP During 1981 and the 1981-1982 Refueling Outage

Dear Sir:

Enclosures (1) and (2) are submitted for your review in accordance with 10CFR50.59 requirements. These enclosures contain summaries of modifications, changes, or experiments installed or implemented at the James A. FitzPatrick Nuclear Power Plant in 1981 and during the 1981 refueling outage which ended on March 10, 1982.

If you have any questions concerning these reports, please contact Mr. Victor M. Walz at (315) 342-3840, extension 265.

Very truly yours,

Corbin A. McNeill, Jr.
Resident Manager

CAM:VMW:nvw
Enclosures (2)

cc: USNRC, Director, Office of Inspection and Enforcement
Internal Distribution

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REPORT OF PLANT MODIFICATIONS, CHANGES, AND EXPERIMENTS IMPLEMENTED DURING 1981 AND THE 1981/82 REFUELING OUTAGE

F1-77-28 This modification extends the fire suppression system into the new grease and oil storage rooms. This improvement is necessary for adequate fire protection in that area.

F1-79-13 This modification consists of installing interlocks between the Relay Room Cardox System and the Relay Room Ventilation System so that the operation of the ventilation system does not reduce the concentration of the carbon dioxide fire suppressant during a fire. When the cardox system is activated, the isolation is accomplished by: a) shutting down the Relay Room supply fans; b) closing the exhaust dampers which shut down the exhaust fans; and c) during the initial CO2 discharge only, leaving the exhaust damper (70MOD 102B) open to allow displaced air to escape.

This modification also consists of installing conduit and cabling between the exhaust fans and the CO2 fire suppression control panels for the Cable Spreading Room, the North and South Cable Run Rooms, and the Emergency Diesel Generator Rooms. The control circuits for each exhaust fan will be modified so that the fan will be shutdown automatically upon the initiation of its respective CO2 system, thereby preventing air dilution of the CO2 concentration. This modification will improve the effectiveness of CO2 fire suppression systems and does not alter any part of the FSAR. No changes to the assumptions used in the FSAR or other safety analysis reports will result. The results of the accident analysis will be unchanged, and the safety margins as defined in the bases of the Technical Specifications will not be reduced.

F1-79-16 This modification modified pipe supports in accordance with the pipe stress reanalysis required in Bulletin 79-02/14.

F1-79-18 This modification installed break flanges on various safety or relief valves. The present piping arrangements did not use flanges and consequently have inlet and outlet piping permanently welded to the subject valves.

Installation of the flanges in piping systems does not alter the original function of the piping system, nor does installation of flanges in any way degrade the performance capabilities of the piping system. Originally, many piping systems are designed with as few welds as possible to avoid excessive NDE work. As such, butt welding reduces the number of welds required by one-half. However, those items which require relatively frequent removal from piping systems for maintenance or testing are usually provided with break flanges. This modification installed these necessary break flanges which should have been used in the original design. In essence, the installation of break flanges does not in any way affect the original piping system. Flanges are approved for use by ANSI B31.1 and does not constitute a change in original design.

These flanges were installed in piping systems which have been designed to withstand the design basis earthquake (SSE). The added concentrated mass loads of 4 to 80 pounds will not affect the seismic capacity of the attached piping due to the relatively small magnitude of the added loads. For this reason, the flanged installation does not require a seismic re-analysis to verify seismic adequacy.

The FSAR, Technical Specifications, NRC criteria and other design criteria have been reviewed for this plant and find no circumstances where this modification will violate any existing criteria.

F1-79-24 This modification consists of installing additional fire detectors (ionization and ultraviolet types) throughout safety-related areas of the plant. Also, the signal initiating and sounding circuits will have Class A supervision. All fire doors not already supervised by the security system will be electrically supervised. Also, for personnel safety, a system allowing the inhibiting of local cardox system actuation (with appropriate alarming) will be installed on a new main fire protection panel in the control room.

F1-79-24
(Cont'd)

This modification will enhance the fire detection capabilities of the Fire Protection System and does not alter any part of the FSAR. No changes to the assumptions used in the FSAR or other safety analysis reports will result. The results of the accident analysis will be unchanged, and the safety margins as defined in the basis of the Technical Specifications will not be reduced.

F1-79-25

This modification converts the existing manually actuated water spray systems to automatically actuated fire suppression sytems (water spray or foam) and provides rated fire barriers around the RCIC and HPCI turbines to provide separation.

This modification does not alter any part of the FSAR. No changes to the assumptions used in the FSAR or other safety analysis reports will result. The results of the accident analysis will be unchanged, and the safety margins as defined in the basis of the Technical Specifications will not be reduced.

F1-79-27

This modification consists of installing fireproofing material on the supporting structural steel between the electrical switchgear bays, between the emergency diesel generator rooms, around the fire foam room, the RCIC enclosure and the 15MOV-175B enclosure.

F1-80-07

This modification consists of upgrading the electrical cable penetration seals to provide 3-hours of fire protection. The upgrading will be performed using existing procedures for Tech-Sil RTV Silicone Foam and Elastomer or new procedures written for fire stop material presently being tested and qualified.

This modification will improve the integrity of the fire barriers, improving the fire protection within the plant and does not alter any part of the FSAR. No changes to the assumptions used in the FSAR or other safety analysis reports will result. The results of the accident analysis will be unchanged, and the safety margins as defined in the basis of the Technical Specification will not be reduced.

F1-80-08 This modification consists of sealing pipe penetrations through fire barriers between safety related fire areas. Various methods and materials (i.e., boots, elastomer, high density gel, foam) will be used depending on the temperature of the pipe, the desired amount of radiation shielding, and whether the penetration is to be air tight.

F1-80-16 This modification installed a high range gamma radiation monitor system at the James A. FitzPatrick Nuclear Power Plant (JAFNPP) to monitor gamma radiation in the primary containment after a postulated accident. The system will detect and measure radiation levels from 10^0 to 10^8 R/hr within the reactor containment and generate an analog electric current ranging from 10^{-11} to 10^{-3}A. The purpose of the installation is to improve the ability to assess and possibly control the course of an accident by providing information related to the extent of core damage that has occurred or may occur during an accident.

The installation sequence has been reviewed for impact on the operation of the plant. No unreviewed safety question has been presented, and the probability of an accident or malfunction has not been created.

F1-80-18 This modification entails the upgrading of the electrical facilities in the Radiological Counting Laboratory by providing a redundant, reliable source of power from the emergency bus.

F1-80-24 This modification consisted of rerouting approximately 100 safety related circuits within conduit to provide adequate divisional and fire separation.

The end result of this modification will be to improve the reliability of each nuclear safety systems and engineered safeguard systems as described in Safe Shutdown Analysis for the James A. FitzPatrick Nuclear Power Plant.

F1-80-25 This modification consists of: a) The installation of a sectional concrete hatch cover over the open hatch in the northwest corner of the reactor building at elevation 300'-0", b) construction of a three hour fire rated enclosure around the circular stairwell at elevation 272'-0", c) construction of a 9' high three hour fire rated wall between motor control centers C-142/C-161 and C-132/C-151 and associated tray water spray, d) providing fire separation between the two redundant emergency service water valves MOV-175A and B by enclosing MOV-175B in a three hour rated steel enclosure, and e) installation of two fire dampers in the ductwork in the southwest corner of the reactor building at elevation 272.

This modification will enhance the fire protection of the reactor building by providing for additional fire barriers. It will provide additional protection between MCC's C-141/C-161 and C-132/C-151 and to the cable trays and conduits above. It will enhance the independence of the emergency service water loop by protecting the control valve of one loop in case of an unmitigated fire in that area. Vents with fire dampers will reduce heat buildup. The enclosure panels are designed to be removable to facilitate service. This modification does not alter any part of the FSAR. No changes to the assumptions used in the FSAR or other safety analysis reports will result. The results of the accident analysis will be unchanged, and the safety margins as defined in the basis of the Technical Specifications will not be reduced.

F1-80-28 This modification modified the Primary Containment Purge System to insure the capability for purge initiation without reactor building entry. Also prevent single failure problem (NUREG-0578, 2.1.5a). The fail safe logic for operation of the vent and purge bypass valves is comprised of DC and AC power supplies in such a combination that no single failure can prevent purging or cause loss of containment integrity from either the drywell or from the torus.

F1-80-28
(Cont'd)

Since the primary reason for the addition of the motor operated valves is to provide a fail safe purge path from the containment under any accident conditions - the last paragraph of FSAR Section 5.2.3.7 Primary Containment Cooling and Ventilation Systems, shall be expanded to explain the use of the bypass valves. Table 5.2-2, Sheet 3 of the FSAR (Primary Containment System Principal Penetrations and Associated Isolation Valves) and Table 3.7-1 of the Technical Specifications (Process Pipeline Penetrating Primary Containment) are to be revised to indicate the additional valves. Finally, the installation sequence has been reviewed for tentative impact on the operation of the plant. No unreviewed safety question has been presented and the probability of an accident or malfunction has not been created.

F1-80-32

This modification consists of installing a 32 window spare annunciator assembly into the 09-4 panel and wiring it into the 0943 panel in the Relay Room.

F1-80-34

This modification consists of adding one additional relay contact in the HPCI Gland Seal Exhauster circuitry to prevent the HPCI gland seal exhauster from stopping automatically when the initiating signal is reset. The revised control circuitry will require direct operator action in order to stop the HPCI Gland Seal Exhauster after initating signal reset.

The reset of the engineered safeguard is not addressed in the FSAR. This change will enhance safety because manual operator action must be taken in order to secure the HPCI Gland Seal Exhauster motor when the HPCI System function is no longer necessary to maintain reactor water level.

The installation sequence has been reviewed for impact on the operation of the plant. No unreviewed safety question has been presented, and the probability of an accident or malfunction has not been created.

F1-81-01 This modification consists of the installation of four pressure switches in the CRD Control Air System to signal low air pressure and to initiate a reactor SCRAM when the air pressure drops below 50 psig.

The four pressure switches will be wired in series with the SIV High Level Switches to the two trip channels of the RPS and will initiate the scram sequence on the basis of one-out-of-two-taken-twice logic. The pressure switches will be individually connected to the air header. During the wiring of the pressure switches a half scram condition will exist. This will not preclude a safety shutdown of the plant should a signal to the other scram channel be generated.

F1-81-02 This modification changes logic for drywell equipment and floor drain isolation valves (20AOV-83, 20AOV-95) such that when containment isolation logic is reset, valves will not automatically re-open. Modification will require deliberate operator action to open valves after isolation signal is cleared. Switches must be taken to close, isolation reset, then valves may be opened.

This modification will enhance the reliability and does not alter any part of the FSAR. No changes to the assumptions used in the FSAR or other safety analysis reports will result. The results of the accident analysis will be unchanged, and the safety margins as defined in the bases of the Technical Specifications will not be reduced.

F1-81-03 The RCIC system was modified to incorporate an automatic reset on high reactor vessel water level. The RCIC system would then restart on low water level. The operator would then only have to verify proper operation.

The installation sequence has been reviewed for impact on the operation of the plant. No unreviewed safety question has been presented, and probability of an accident or malfunction has not been created.

F1-81-04 A spare electrical feeder from 600 VAC load center (L24) was installed to supply power to the maintenance garage, accounting office, and other temporary electrical services as required in the east area of the plant. Spare breaker in compartment 3A of L24 (BUS 12400) is utilized with 225 Amp, EC-1 series trip devices. This new breaker feeds two 600 /208Y air-cooled distribution transformers 71TBT-3A-3B located in the Maintenance Garage. 71TBT-3A, 3B supply distribution panels in the various buildings in the east area of the plant.

F1-81-05 The purpose of this modification is to automatically switch the RCIC pump suction from the condensate storage tanks to the suppression chamber when the condensate storage tanks reach a predetermined low level. In order to do this, new level switches shall be installed adjacent to existing HPCI level switches, 23LS-74A, 74B, 75A and 75B. These new switches shall energize relays in panel 0930, which in turn shall operate valves 13MOV-39 and 13MOV-41, allowing suction to be taken from the suppression chamber. Valve 13MOV-18 will close automatically once valves 13MOV-39 and 13MOV-41 are fully opened, preventing suction from the condensate storage tanks. Two new annunciation windows will be provided to indicate low level in the condensate storage tanks.

The installation sequence has been reviewed for impact on the operation of the plant. No unreviewed safety question has been presented, and the probability of an accident or malfunction has not been created.

F1-81-06 Modifications to upgrade the 8" block walls were performed to meet a 3 hour fire rated construction as required by SER Appendix R and NRC I.E. Bulletin 80-11.

a. The following wall sections were upgraded:

Electrical Bays - EB-272-10, 11, 12, 15, 16 (both sides)

Cable Tunnels - CT-286-4 & 5 (both sides)

<u>Modification Procedure</u>:

F1-81-06
(Cont'd)

-Attach 3.4 lb. galvanized wire lath by means of lathing nails.

-Coat with a 1:2 1/2 ratio of Portland cement and sand mixture, 3/4 inch thick, water added to mix as needed to produce a workable consistency.

-Maintain existing control joints.

The additional weight on the walls due to the fireproofing does not produce an unresolved safety question. This was addressed in a re-evaluation of the masonry wall design.

Due to the minor scope of the modification and administrative controls, the modification installation does not result in an unreviewed safety question.

F1-81-07

This modification changes the electrical power supply of one of the two reactor vessel wide range level indicators (02-3 LI-85A, B). The existing power supply for both wide range (Yarway) level indicators is the 120 VAC Uninterruptible Power System (UPS) with both indicators being fed from distribution panel 71ACUPS-1. This modification changes the power supply for one of these indicators (02-3LI-85B) to the "B" AC Emergency Power System (Blue) with its source being 120 VAC distribution panel 71ACB2 in the Relay Room.

F1-81-08

This modification consists of the addition of new pipe supports, modifies some existing supports and the deletion of other supports of the Scram Discharge Volume (SDV) headers, the Scram Instrument Volume (SIV), the 2" SDV drain lines to the SIV, the SDV vent lines up to and including the vent valves and the SIV drain line up to and including the drain valve.

It is concluded that for any of the above cases the reactor can be shutdown safely and the leakage of reactor coolant uncontrolled. Strict administrative controls and precautions stated in the installation instructions will minimize the possibility damage to the CRD hydraulic system.

Where the installation of revised or new supports affects the loading of adjacent supports or piping, an installation sequence is included on the installation sketches.

F1-81-08
(Cont'd)

The new seismic analysis and design of modified or new supports meets the original design criteria addressed in the FSAR and original design specifications.

F1-81-09

This item modified the following masonry wall panels:

1) EB-272-3, 5, 6 & 11
2) ACE-300-2
3) MCR-300-2, 3 & 8
4) EGB-272-1, 2, 3, 4, 5 & 8
5) CT-286-4 & 5

The modifications will keep the plant in compliance with present commitments for concrete masonry walls. Safety margins for accident analysis will not be degraded as the modifications will improve the ability of masonry walls to perform their design functions.

F1-81-10

The Reactor Protection System Power Supply protective circuitry shall be upgraded by installing eight (8) Electrical Protection Assemblies (EPA), each consisting of a seismically qualified circuit breaker with protective circuits for undervoltage, overvoltage, and underfrequency, packaged in an enclosure designed to be wall mounted. Two of these assemblies will be connected in series, between each power source and the RPS bus. The enclosures will be mounted separately from the MG sets, and separate from each other. Each EPA consists of trip components which interrupt the input power whenever voltage or frequency exceeds their normal tolerances. If one of the new EPA units is switched to the maintenance mode for calibration purposes annunciation is provided in the Control Room. The original RPS MG Set overvoltage, undervoltage and underfrequency relays are retained for annunciation purposes only. The original output molded case circuit breaker consisted of an integral undervoltage shunt trip. This breaker is being replaced with an identical breaker without the shunt trip. Trouble annunciation of each RPS MG Set is provided from these three (3) relay devices.

F1-81-12 This modification consists of the changing of existing pipe supports BFSK-659-H6 and BFSK-658-H4 to the modified pipe supports BFSK-701 and BFSK-707 on the fire protection water line to transfer the lateral pipe support loads from masonry walls PH-255-6 and PH-255-3 to the floor through use of knee braces.

It is concluded that if the fire protection water line is accidently damaged during modification of the fire protection water pipe supports the establishment of a backup fire suppression water system and a continuous fire watch will provide sufficient fire protection. Removing the pipe loading from the masonry walls enhances the capability of these walls to withstand design base earthquake loading.

F1-81-13 The existing A-concentrator will be replaced with a unit fabricated from Incoloy 825. This will increase the corrosion resistance of the concentrator which is presently fabricated from 300series stainless steel. In addition to this material change, a lower demister spray will be added to improve operation and several sample lines enlarged to prevent plugging.

This modification will have no effect on the evaluation of the plant's environmental impact. All safety questions identified have been addressed.

F1-81-14 This modification consists of the addition of two new relief valves branching off isolation valve CAD-40 of the nitrogen pneumatic supply system to the Main Steam Isolation Valves (SMIV) and the Main Steam Safety Relief Valves (SRV)*. Also included in this modification is the rewiring of pressure switch 27PS-100. The existing pressure switch will be wired and set to signal an alarm at both low and high pressure settings. The computer alarm will be changed to correspond to the dual function.

It can be concluded that the malfunction of any component of this modification does not effect the safety of the systems involved.

F1-81-17 This modification consists of the changing of the internal valve trim in the low flow feedwater control valve (34FCV-1370) from linear to equal percentage trim.

The major safety concern of this modification is low reactor vessel water level due to malfunction of the low flow feedwater control.

If the reactor vessel water level is determined to be low, due to failure of the flow control valve, the opening of 34MOV-100a or 34MOV-100b will provide flow to return the reactor vessel water to the required level.

Per FSAR, Section 14.5.4.3 the loss of feedwater event has been evaluated and this event does not result in an unreviewed safety question.

It can be concluded that if the low flow feedwater control does malfunction the reactor vessel water level can safely be maintained.

A special procedure will be followed during the installation of the new valve trim to preserve cleanliness of the feedwater system.

F1-81-18 To prevent flying debris from entering vent and purge piping, and thereby preventing containment isolation valves from closing subsequent to a loss of coolant accident, six (6) grating covers were installed on the inside of penetrations X-25, 26A, 26B, 71, 205 and 220.

The design basis for the sizing of the grating was for the purpose of preventing loose material from entering the penetration. Additionally, it is sized so that the reduction of the effective cross sectional area of the opening to the drywell would not prevent venting the containment, Post-LOCA. Stone & Webster has designed the grating to be capable of withstanding dynamic LOCA forces. Seismic loadings were found to be significantly less than the dynamic LOCA loads and were discounted in this design. The Authority has added a specific checkoff to OP-65 to check the grating covers for blockage.

F1-81-19 Control circuits on those primary containment isolation valves which are required for venting and purging were revised to allow the circuitry isolation permissives to be individually overridden. This allows the valves to be manually opened with an isolation signal present and yet automatically close if another isolation signal is initiated.

Affected valves:

PCP Exhaust Isolation Bypass Valves - 27MOV-113, 117, 122 and 123

Drywell Inerting & Purge Supply Containment Isolation Valves - 27AOV-111, 112, 113 and 114

Suppression Chamber Inerting and Purge Supply Containment Isolation Valves - 27AOV-115, 116, 117, 118

The control circuit modification is achieved through the addition of three (3) keylock override switches per redundant channel. The net effect is one override switch per each of the four isolation signals (low reactor water level, high drywell pressure, reactor building HVAC high radiation, containment high radiation). The contacts of the switches and the isolation initiation permissives are multiplied through the use of control relays in such a fashion that the relay contact used in the valve control circuits represents the isolation signal and the override signal. The valve actuation circuit then has four relay contacts representing each of the four isolation permissives and respective override switch.

F1-81-22 This modification consists of the addition of new pipe supports, modification of some existing supports, and the deletion of other supports from the CRD insert and withdrawal lines. This modification effort will take place during the 1981 Refuel Outage.

The new siesmic analysis and design of modified or new supports meets the original design criteria addressed in the FSAR and original design specifications.

F1-81-23 An enclosure/plug slab structure, with penetrations across its thickness, is provided, replacing the present one made of solid concrete.

The enclosure is part of the reactor building and during the outage will serve as a primary containment.

The enclosure will be designed to stand the impact of a tornado missle as defined in JAFNPP FSAR Section 12, page 12.4.1.

The implementation of this modification does not constitute an unreviewed safety question pursuant to 10CFR50.59.

F1-81-24 This modification is the second phase of modifications associated with the torus. The following design modifications will be made during the 1981 Refueling Outage:

1) Install safety relief valve "T" quenchers after removing the existing piping and ramshead supports. (These are fabricated by General Electric.)

2) Fabricate and install SRV "T" quencher supports.

3) Procure and install SRV discharge line vacuum breaker valves in the drywell.

4) Replace the RHR return lines inside the torus with new piping and a pipe support.

5) Perform various structural modifications to the torus interior catwalk to strengthen it or remove sections.

6) Install gussets to stiffen the intersection of the vent header downcomers.

7) Fabricate and install additional supports on the torus spray header.

8) Install thermowells in the torus shell.

9) Replace a fabricated elbow on the HPCI line with a radius elbow and additional piping.

F1-81-24
(Cont'd)

10) Replace a section of the RCIC line in the torus with new piping. Three pipe supports will be fabricated and installed on this line.

11) Remove and replace an existing manhole cover in the torus shell used for electrical connections.

The short term program verified that the Mark I containment system would maintain its integrity and functional capability when subjected to loads induced by a postulated design-basis LOCA and verified that the plant would operate safely while the comprehensive long term program was being conducted.

The long term program objective has been to establish design basis loads that are appropriate for the anticipated life of the Mark I facility and to restore the design-safety margin. Generic methods have been developed to define suppression pool hydrodynamic loading situations and techniques for structural assessment of the problem. The generic analysis techniques have been used to perform a plant unique analysis for the JAF plant. This analysis and subsequent design changes included in this modification provides a configuration that restores the original design-safety margin to the primary containment.

The Owner's Specification delineates the specific design, fabrication installation, and NDE requirements. The use of ASME Boiler and Pressure Vessel Code, Section III, 1977 Edition through Summer 1978 Addenda in lieu of the original construction code, ANSI B31.1 (1969 Edition) for replacements is permitted since the following requirements have been met:

a) The requirements affecting the design, fabrication, and examination of the replacement are reconciled with the Owner's Specification.

b) Mechanical interfaces, fits, and tolerances that provide satisfactory performance are not changed by using the later code.

F1-81-24 (Cont'd)

c) Modified or altered designs are reconciled with the Owner's Specification through GE topical reports "Containment Design Rules and Classification" (NEDO-24522) and "Mark I Containment Program Structural Acceptance Criteria Guidelines for Containment System Modification" (NEDO-24629).

The construction NDE and test requirements which are specified in the Owner's Specification are equal to or better than the original construction requirements.

Materials are compatible with the installation and system requirements. Therefore, the later code requirements are equal to or better than the original FSAR design criteria.

F1-81-25

Provide additional isolation valves in the Reactor Building Closed Loop Cooling System to enable maintenance with minimum system effects.

F1-81-47

This modification involved the installation of an entirely new system for monitoring the suppression pool temperature in each torus bay and the calculation and display of the overall bulk suppression pool temperature. The bulk temperature is calculated by electronically averaging the individual torus bay temperatures. Paragraph 5.2.3.10 of the FSAR provides a description of the existing suppression pool temperature monitoring system. The new system will supplement the existing system and will provide the primary source of bulk pool temperature indication. This paragraph will be expanded to also describe the new bulk temperature system.

Table 3.2.6 in the JAFNPP Technical Specification which lists the required plant surveillance instrumentation must be updated to reflect the new instrument channel provided by this modification. The range of this new channel (30°-230°F) must also be indicated. The conditions for operation as a result of the loss of an insrument channel as discussed in this table must also be updated to reflect this modification.

A review of the FSAR, the plant Technical Specifications, and the NRC safety evaluation leads to the conclusion that this sytem enhances the operator's ability to ensure that the suppression pool temperature is within and will remain within, the allowable limits set forth in the plant Technical Specifications.

F1-81-48 This modification involved switching the power supplies for outboard MSIV isolation logic as follows:

		Power Supplies	
		Existing	Post-Modification
1.	Trip Logic "A"	RPS "B"	Battery "B"
2.	Trip Logic "B"	Battery "B"	RPS "B"

The modification was accomplished by reterminating two existing cables in Panel 09-42 (PCIS Cabinet).

The changes required by this modification affect only the outboard MSIV isolation logic ·ˌer supplies. No other changes to the .board MSIV's or the Reactor Protection System are to be made.

This modification will eliminate potentially severe reactor transients resulting from unnecessary main steam line isolations due to the loss of one RPS power supply.

F1-81-49 To change the operation of Torus to Drywell pressure instrument isolation valves from ganged to individua. operation, two control switches and two relays will be added to the Primary Containment Purge Panel (27 PCP). Valves 16-1AOV-101A & B and 16-1AOV-102A & B will be operated by four individual switches. Currently one switch control 16-1AOV-101A and 16-1AOV-102A which are on two separate lines. Another switch control 16-1AOV-101B and 16-1AOV-102B which are also on separate lines.

To change Automatic Isolation from Fuel Pool Vent High Radiation Monitors (17RIS-456A & B) to Reactor Building Vent High Radiation Monitors (17RIS-452A & B). This requires a minor wiring change at AR6-A and AR6-B.

The Technical Specification, NRC Regulatory Guides and the FSAR were researched to determine if this modification would effect any system design basis, increase the possibility of safety hazards, or increase the possibility of accidents. This modification does not depart from previous design bases, nor increase the possibility of any safety hazards or accidents. This modification does not affect containment integrity:

F1-81-49
(Cont'd)

1) The remote manual operation of the valves is only being modified from ganged to individual.

2) Automatic operation is modified to isolate on the same signal as all other Group B containment isolation valves.

3) None of the mechanical aspects of these valves have been modified.

F1-81-50

The proposed modification involves the cutting and removal of existing core spray austenitic 304 - stainless steel piping (with a maximum carbon content of .08%) from the safe-ends to the manual block valves. This piping would be replaced with type 316 stainless steel with a maximum carbon content of .02%. This alloy has been shown to be resistant to sensitization by welding, and therefore less prone to inter-granular stress corrosion cracking.

Since the pipe installation represents as closely as possible a one for replacement, and since the new material is designed to meet or exceed the original design requirements, implementation of this modification does not constitute an unreviewed safety question pursuant to 10CFR50.59.

F1-81-51

This modification replaces existing valve ESW-23 in the Emergency Service Water System with a functionally identical replacement valve. The physical performance characteristics of the new valve are identical to those of the current ESW-23 valve.

The FSAR, Technical Specifications, NRC criteria and other design criteria for this plant have been reviewed and there are no circumstances where this modification will violate any existing criteria.

SUMMARY OF TESTS AND MINOR PLANT MODIFICATIONS IMPLEMENTED AT JAFNPP DURING 1981 AND THE 1981-1982 REFUELING OUTAGE

Evaluation No.	Summary
JAF-SE-81-001	The repair entails the on-line sealing of the testable check valve, 13-AOV-22, located in the RCIC pump discharge line between the motor operated isolation valve, 13-MOV-21, and the "A" feedwater line. Due to a leaking seal, leakage has developed from the valve leakoff port. The repair of the valve involves the seal mechanism and <u>does not</u> effect the integrity of the valve seal.
JAF-SE-81-002	The purpose of this preoperational test procedure is to demonstrate the proper operation of the interlocks between the Relay Room Cardox System and the Relay Room Ventilation System so that the operation of the ventilation system does not reduce the concentration of the carbon dioxide fire suppressant during a fire. When the cardox system is activated, the isolation of the Relay Room is accomplished by: a) shutting down the Relay Room supply fans; b) closing exhaust fan discharge damper 70MOD-102A which shuts down the exhaust fan FN-13A; c) shutting down the exhaust fan FN-13B; and d) exhaust fan discharge damper 70MOD-102B may be opened manually during the initial CO2 discharge to allow displaced air to escape. After CO2 discharge, 70MOD-102B may be manually closed. These interlocks were installed under Plant Modification Number F1-79-13B. The conduct of this preoperational test does not alter any part of the FSAR. No changes to the assumptions used in the FSAR or other safety analysis reports will result. The results of the accident analysis will be unchanged, and the safety margins as defined in the bases of the Technical Specifications will not be reduced.

JAF-SE-81-004 The work consisted of hoisting new fuel channel crates from the Reactor Building Track Bay, Elevation 272' to the Refueling Floor, Elevation 396'-6". The crates are 3 foot by 4 foot by 14'-6" long and weighs 2500 lbs. and are considered a heavy load being lifted over safety related equipment.

Because of the geometry of the hoist area and size of the fuel containers, it is concluded that a fuel channel crate cannot fall onto the small corner area where there is a potential for structural damage.

JAF-SE-81-006 This test involves decreasing load limit setpoint until turbine bypass valves open, observation of valve stability, and then restoration of load limit to normal. Following this, if oscillations cease, the pressure regulators will be swapped to B in service and stability again observed.

Based upon GE analysis, the probability and consequences of an accident or malfunction described in the FSAR and other safety evaluation reports is not increased. The probability of an accident or malfunction not analyzed is not created. The margin of safety defined by Technical Specifications is not reduced. Therefore, this test is not an unreviewed safety question pursuant to 10CFR50.59.

JAF-SE-81-007 Replace torus narrow range level transmitter (23-LT-201A) with an equal to or better than transmitter supplied by a different manufacturer. The original instrument has an unknown status of qualification. The replacement transmitter does not experience harsh environments but must function on its normal environment prior to potential accidents but not during these accidents or after any accident.

The replacement GEMAC 555 transmitter meets the performance requirements of the narrow range torus water level application and is acceptable based on performance considerations.

JAF-SE-81-013	Lift a disposal liner for the Chem-Nuclear Cask 4-45 from the Reactor Building 272' Track Bay to the Reactor Building 369' elevation. Place it in the Spent Fuel Pool and load it with cut-up LPRMs for ultimate removal. This work shall be accomplished cautiously with preventive measures and administrative controls taken to preclude damage to any existing equipment/components.
JAF-SE-81-017	Lift the General Electric fuel containers with unirradiated fuel from the Reactor Building 272' Track Bay to the Reactor Building 369' elevation. Full containers will be located in the designated storage area for fuel inspection and channeling at a future date. Due to the minor scope of the intended maintenance activity and administrative controls, this maintenance task does not result in an unreviewed safety question.
JAF-SE-81-022	This modification involves the construction of a poured, reinforced concrete wall approximately thirteen (13) inches thick and to a height of fourteen (14) feet six (6) inches above grade. The CST Shield Wall is to be located outside of the existing CST protective wall.
JAF-SE-81-023	The scope of the Revision of Preoperational Test 76R is to fabricate and temporarily install a light-weight, 28 gauge 3' long x 3' wide x 3' high section of suction duct to RCIC Enclosure Backdraft Damper Fan. The purpose of the duct is to ascertain actual installed performance of fan by physical measurement. (Pitot tube traverse or anemotherm.) Buffalo Forge sent a field technician to JAF site (to do this measurement). They determined why their fan did not open damper louvers over 10^{o} and made recommendations for a new design to remedy the situation. The redesign will be QA Category I - seismically evaluated to match all other equipment. Upon evaluation of temporary duct for measurement purposes, we did not design it per seismic qualifications, but for light-weight, and ease of handling. However, it will only be installed for a very short period of time during actual measurement - on the order of 4 hours.

JAF-SE-81-923 (Cont'd)

There are no extra safety hazards associated with this installation which are not present on any other modification or which could result in a failure, mechanical, electrical, or otherwise. There is no other design consideration thought of at this time which could prevent or mitigate failures; on the contrary, the light-weight of the temporary duct would be a plus inside RCIC enclosure in case of a seismic event, whereas it would create less hazard to breaking surrounding equipment and less hazard to personnel present at time of testing.

Neither the implementation of this temporary modification or performance of this test nor the subsequent installation of a permanent design derived from this temporary modification constitutes an unreviewed safety question pursuant to 10CFR50.59.

JAF-SE-81-024 Separate the feed piping to A & B concentrators so that one can be run while the other is out of service. In addition, split the feeds from the waste neutralizer tanks so that each can be processed independently. This modification does not alter the agility of the radioactive waste building to contain liquid during a seismic event or under normal operations. Therefore, it does not increase the probability of occurrence or consequences of an accident or malfunction of equipment previously evaluated in the FSAR. It will increase the capabilities of the Radwaste System by allowing operation of one concentrator while the other is isolated for Boil-Out.

JAF-SE-81-025 Reroute concentrator feed piping to prevent line plugging. This modification does not alter the ability of the radioactive waste building to contain liquid during a seismic event or under normal operations. Therefore, it does not increase the probability of occurrence or consequences of an accident or malfunction of equipment previously evaluated in the FSAR. It will increase the availability of the radwaste system by allowing concentrator operation with less line plugging.

JAF-SE-81-031 This test involved the recording of the startup flow rate, through the low flow feedwater control by use of an ultrasonic flowmeter to determine the flow bias setting

JAF-SE-81-031
(Cont'd)

for the minimum flow rate of 100 gpm and to verify flow control characteristics with the modified flow control valve. Flow measurements will be taken from the time of initial opening of low feedwater flow control valve (34FCV-173) to the time of opening of the main feedwater flow control valves.

The major safety concern during this test is low reactor vessel water level due to malfunction of the low flow feedwater control. If the reactor vessel water level is determined to be low due to failure of the feedwater control valve, the opening of motor operated valves 34MOV-100 A & B will provide sufficient flow to return the reactor vessel water to the required level.

Per FSAR section 14.5.4.3, the loss of feedwater event has been evaluated and this event does not result in an unreviewed safety question.

It can be concluded that if the low flow feedwater control does malfunction, the reactor vessel water level can safely be maintained.

JAF-SE-81-033

This safety evaluation addresses the repair to an electrical conductor and associated conduit which was damaged during the removal of ureathane foam fire barrier material at wall sleeve. Cable number 1RPSCUC158 which is a single conductor stranded size 10 wire had its insulating jacket perforated and a short circuit occurred which damaged the wire conductor to the extent that the wire had to be spliced. The location of the damage was such taht not enough slack in the cable could be gained to effect a repair.

A new conductor will be installed and a splice made in the "C" conduit fitting about 10 feet from the wall on conduit 1CC001UB4. A second splice shall be made at JB RPS15. The replacement conductor shall be comprised of one of the conductors of a two conductor, 600 volt cable. The reason for use of the two conductor cable is that none of the single conductor #10 AWG is readily available.

JAF-SE-81-036 RHR service water pumps 10-P-1A and C will be started individually and cooling water flow observed for each. Each pump will then be individually stopped and examined for respective cooling water flow stoppage. Pumps will be examined for smooth operation. The plant will be in shutdown condition for this test.

As a result of Modification F1-80-24 (PF-11), new cable rerouting for control power to 10SOV-101A and C requires testing these valves for proper operation.

In accordance with the FSAR, Section 1.6.2.18, Residual Heat Removal Service Water Syste, the results of the accident analysis are unchanged by operability test 10A to which a copy of this nuclear safety analysis is attached. "The system consists of two independent loops, each with two pumps and the associated valves and piping." Operabi ity Test 10A affects only the A loop.

The safety margins defined in the bases of the Technical Specifications are not reduced. As stated on page 127 of the bases, any two of the RHR service water pumps can satisfy the cooling requirements of the RHR system. Operability test 10A affects only two of the four RHR service water pumps.

JAF-SE-81-038 Relocated the 1) CNS 4-45 Cask Liner, 2) Atcor Crusher/Shear Unit #3 Liner, 3) Chem-Nuclear Disposable Container from their present location in the Spent Fuel Pool by moving them south approximately 15' to a new location in SE corner of Fuel Pool directly under the existing work table.

Relocated all three units out of the way to expedite removal of old spent fuel racks and installation of new high density spent fuel racks. This work shall be accomplished cautiously with preventive measures and administrative controls taken to preclude damage to any existing equipment/components.

JAF-SE-81-039 Removed existing CRD temperature recorder and installed new computerized data acquisition system.

Equipment temperature monitors of CRD mechanism alert operator of overheating conditions which over a long term would result in seal degradation. Safety related functions are not performed by this system.

JAF-SE-81-041 Installation of one 4" diameter rigid steel conduit from Telephone PBX room to east wall of Turbine Building at elevation 272'-0". Conduit shall be used to route 400 pair telco cable to provide future telephone service to trailers/buildings in East lot.

JAF-SE-81-042 Provide potable water, supply line for sprinkler system, septic system, A.C. power supply, gaitronics connections, and security signal coax cable, for the new secondary access point. The only part of this modification which is Class 2A is security signal coax. The microwave cans and fence shaker alarms will be shutdown in the construction area and a guard will be provided. All other work in modification is non-safety related.

JAF-SE-81-045 This safety evaluation addresses the change in tolerances for the Reactor High/Low Water Level Local Indicating Switches 02-3-LIS-101A-D. The instruments are Barton Type 288A, with a range of 0-60 inches of water and a required accuracy of ± 1% full scale (-0.6 inch of water column) for points past switch actuation, as published in General Electric's Instrument Data Sheet No. 234A9301RK, Sheet 15. Switch actuation causes interference with indicator accuracy, which results in out-of-tolerance indicator calibrations.

To reduce this out-of-tolerance condition and continuous Occurrence Reports, the tolerance was increased from the present procedural tolerance (F-ISP-3) of ± 1% (0.6 in W.C.) to ± 3% full scale (+1.8 in. W.C.) for the indicator only.

The setpoints and tolerances of the high (≤ 58 inches of water) and low (≥12.5 inches of water) trip points, which are Technical Specification requirements, will not be affected by this change. The results of this evaluation indicate that the installation of the Cable Tunnel Masonry wall modifications will not result in an unreviewed safety question due to proper administrative controls and redundant systems.

JAF-SE-81-046 This test is intended to verify proper operation of various motor operated valves of the Residual Heat Removal System 10, Loop B. This test is required as a result of rerouting control cables to these valves as per Modification F1-80-024 (Fire Hazard Analysis - PF-11).

The valves to be tested are as follows:

1) 10MOV-13B RHR Pump B Suction Valve
2) 10MOV-13D RHR Pump D Suction Valve
3) 10MOV-15B Shutdown Cooling RHR Pump B Suction Valve
4) 10MOV-15D Shutdown Cooling RHR Pump D Suction Valve
5) 10MOV-21B RHR Heat Exchanger Discharge to Suppression Pool Valve
6) 10MOV-34B Suppression Pool Cooling Valve
7) 10MOV-36B RHR Heat Exchanger Discharge to RCIC System Valve
8) 10MOV-38B Suppression Pool Spray Valve
9) 10MOV-39B Suppression Pool Cooling Isolation Valve
10) 10MOV-65B RHR Heat Exchanger Shell Side Inlet Valve
11) 10MOV-66B RHR Heat Exchanger Shell Side Bypass Valve
12) 10MOV-25B LPCI Inboard Valve
13) 10MOV-27B LPCI Outboard Valve

Each valve was tested as signaled from the main control board panel 093 to open and close. The associated red and green light were verified for proper indication of valve operation. In addition, valves 10MOV-25B and 10MOV-27B were tested to be interlocked not to both open when the reactor pressure is high. Since the plant was shutdown for this test, the low reactor pressure signal to valves 10MOV-25B and 10MOV-27B will be interrupted by temporarily lifting the appropriate wire at the 09-3 panel.

The probability of occurrence or consequence of an accident evaluated in the FSAR have not been increased. The possibility of an accident or malfunction of a different type than any previously evaluated has not been created. The margin of safety as defined in the bases of Technical Specification has not been reduced. Based upon the statements made above, the conduct of this operability test (procedure 10B) does not constitute an unreviewed safety question pursuant to 10CFR50.59.

JAF-SE-81-048 Changed the discharge of the condensate receiver tank (87-CR35) vent line, from 20-RV-784B discharge to atmosphere, to sub-cooler E650 ("B" Concentrator) line.

JAF-SE-81-051 This safety evaluation addresses the change in bias settings for the Drywell Equipment and Floor Drain Sump Flow Integrators 20-FQT-530 and 20-FQT-527.

The instruments are General Electric Type 561 flow integrators with a range of 9999.9 x 10 gallons of flow. Bias of the integrator is an adjustment that is made at the zero flow point of the circuit loop. This change does not involve an unreviewed safety question pursuant to 16CFR50.59.

JAF-SE-81-052 Removal of approximately 1250 LF of 10' high chain link fence and posts that are inside our perimeter, and subsequent reuse of approximately 500 LF at new secondary access points.

JAF-SE-81-053 The modification consists of removing of the filler of two sets of sleeves (5" diameter) pass thru cables and hoses needed for outage services, and then after services are completed, remove cables, holes and grout and reseal with fresh grout.

JAF-SE-81-054 A temporary structure made out of 2 seams was placed temporarily over the opening on 300' floor between beams Y and W and 1 and 2. This structure and auxiliary crane was used to lower heavy loads from the 272' floor through the RS.4 opening to the lower level.

JAF-SE-81-055 Temporarily open the east crescent hatch between the crescent and torus room. Two open hatch periods of 36 hours each will be required.

JAF-SE-81-056 This modification consists of changing the existing teflon body O-ring, setscrew and plug with the replacement viton components in solenoid operated valves 27SOV-119A & B through 27SOV-123A & B. A comparison of materials in EPRI RP 1707-3 shows Viton to have a higher threshold of radiation tolerance - 10^5 to 10^6 Rads than Teflon -1.5 x 10^4 Rads.

JAF-SE-81-058 A temporary set of six coolers was installed in the torus room. Cooling water will be taken from the service water system in the west crescent and run through to the torus room via copper piping and firehose and discharged to the west crescent service water return system. This was done by removing Unit Cooler 66-UC-22J from service for the duration of the outage. This is permissible per Technical Specifications 3.11B. After the system was removed, the service water system was restored to its original configuration, inspected and initial service leak tested.

The installation is not safety related and possible failures would not affect safety related equipment.

JAF-SE-81-060 Installed a 600 volt maintenance disconnect switch to provide 50 amp service on elevation 369' of the Reactor Building at column 6 and R-line. Feed shall come from MCC [illegible]51 (RB-326').

JAF-SE-81-061 Installed the Chem-Nuclear System, Inc. Mobile Cement Solidification Unit, to replace CNSI urea formaldehyde system with an approved cement solidification system.

The Chem-Nuclear Mobile Solidification System is a process to solidify liquids and sludges using cement.

JAF-SE-81-062 This modification consists of changing the existing teflon packing with an ethylene propylene O-ring assembly in vacuum breaker valves 27VB-1 through 27VB-5.

A comparison of materials in EPRI RP1707-3 shows ethylene propylene to have a higher threshold of radiation tolerance - 10^6 rads than teflon -1.5×10^4 rads. FSAR Section 5.2.36 states the vacuum breaker capacity is more than adequate (typically by a factor of four) to limit the pressure differential between torus and drywell during post accident drywell cooling conditions.

JAF-SE-81-079 As a result of a Power Authority Safety Evaluation (October 7, 1977) and a Nuclear Regulatory Commission (NRC) Safety Evaluation (April 15, 1981), a voltage profile test must be performed on the JAF Electrical Power Distribution System to verify the original electrical design analysis. The NRC Safety Evaluation requires that the testing be performed under specific guidelines stated in the evaluation. The specific criteria required by the NRC and stated in the subject test procedure are as follows:

a) Installation of temporary instrumentation at various switchgear locations. This includes fuse blocks, terminal blocks, potential transformers, current transformers, and recorders.

b) Loading the station distribution busses, including all Class 1E buses down to the 120/208v level, to at least 30%.

c) Recording the existing grid and Class 1E bus voltages and bus loading down to the 120/208 volt level at steady state conditions and during the starting of both a large Class 1E and non-Class 1E motor (not concurrently). To minimize the number of instrumented locations, (recorders) during the motor starting transient tests, the bus voltages and loading need only be recorded on that string of buses which previously showed the lowest analyzed voltages.

d) Removal of temporary instrumentation.

e) Using the analytical techniques and assumptions of the previous voltage analysis and the measured existing grid voltage and bus loading conditions recorded during conduct of test, a new set of voltages for all Class 1E buses down to the 120/208 volt level will be calculated.

f) The analytically derived voltage values will be compared against the test results.

JAF-SE-81-080 An additional 10,000 CFM of cool air was supplied to the drywell during the Winter '81 outage. This was done by temporarily installing an additional fan and cooler. They were connected to an existing Purge/Ventilation Loop with new temporary duct.

JAF-SE-81-081 Installed a junction box in the run of cable 1VINNNX104 to the seismic sensor 09VD-2 to facilitate future maintenance in a high radiation area. No other changes to equipment setpoints or to operation are involved.

JAF-SE-81-083 Supplied a temporary cross-tie from the suction of the spent resin pump to the suction of the concentrated waste pump to enable transfer of spent resins to solidification rig.

It is concluded that this modification will be installed and operated with radiation exposures as low as is reasonably achievable.

JAF-SE-81-086 Installed approximately 110 ft. of rigid conduit (4" diameter) in the West Cable Tunnel to accommodate a new 400 pair telephone cable required for the expansion of telephone facilities.

Conduit will be seismically supported (seismic class II) to its location in the West Cable Tunnel and its proximity to reactor safety related electrical equipment. The conduit itself is Q.A. Category III. The new conduit will provide the required separation from adjacent safety related cabling.

JAF-SE-81-088 Replaced existing Feedwater Heaters 3A and 3B with new Feedwater Heaters of equal or better quality.

JAF-SE-81-090 HPCI-65, Turbine Exhaust Check Valve, failed its Local Leak Rate Test in accordance with ST-39B. Visual inspection of the valve's internals revealed that the back seat in the valve body was out of flat up to a maximum of 0.006" as measured between the back of the seat ring and the Belzona Molecular Metal.

JAF-SE-81-093 Increased the signal level to Process Computer to improve accuracy and resolution of generator gross, EDIC Line and NMPT Line MWatts and MVars. Relocate computer signal lead from 40MV terminal to 100MV terminal on Watts & Vars Transducers.

This modification does not involve an unreviewed safety question. Process Computer Analog Inputs have no loading effect on signal source; therefore, cannot affect accuracy of the G.E. Type 4701 transducers or data acquisition panel inputs.

JAF-SE-81-095 Erected a vessel decontamination rig and decontaminated the Reactor Pressure Vessel and vessel nozzles starting from the flange to the core spray sparger elevation with the Cleanco Reactor Vessel Decontamination System.

JAF-SE-81-099 Installed a union on the water side, bottom of level transmitter instruments LT-102, 103, 104 and 105 to permit inspection and/or replacement in a minimum amount of time.

JAF-SE-81-102 Changed in input and output bias settings of 02-3-PA-97 (single loop flow amplifier).

The new bias settings do not change the present calibration values. Since the calibration values are not changed, the inactive loop correction remains the same.

JAF-SE-81-103 This safety evaluation addresses the change in calibration values for the Drywell Equipment and Floor Drain Sump Level Transmitters, 20-LT-120A, 120B, 121A and 121B.

The new calibration values will be from 15 to 45 inches for all four (4) transmitters vice the current ranges of 13 1/4 to 43 3/4 inches for 20-LT-120A and B, 12 7/8 to 43 inches for 20-LT-121A, and 15 to 43 inches for 20-LT-121B. (All measurements are from tank bottoms). The level switch setpoints will not be affected.

The purpose of the level transmitters is to provide an output which is used for pump control (20-P-1A/1B and 20-P-5A/5B), valve operations (20-AOV-95), various Hi and Low Level Annunciators and to record sump levels.

JAF-SE-81-103 (Cont'd)

Both drywell equipment and floor drain sumps are 47 3/4 inches deep with the pump suctions extending to within 3 1/2 inches of the tank bottom (1/2 inch including the pump strainer). The level switch settings are at 18, 21, 39 and 42 inches for low-low, low, hi and hi-hi.

With new standardized calibration values of 15 to 45 inches and the level switch settings remaining at present levels, no effect on pump or valve operations will occur. With standardized ranges for all transmitters, calibration of the remainder of the loop will become standard and the probability of calibration error will decrease. This change does not constitute a safety concern.

JAF-SE-81-104 Relocated Radwaste Gates to make control more practical for DAW handling and storage.

JAF-SE-81-105 RCIC-04 and -05, Turbine Exhaust Check Valves, failed their local leak rate test in accordance with ST-39B. Visual inspection of the valves internals revealed that the backseats in the bodies of the valves were lightly pitted. This pitted condition, approximately 0.002" deep, was preventing full contact between the seat rings backs and the valve body back seats.

This modification was to restore full contact between the seat rings backs and the valve body back seats. This was accomplished using Belzona Molecular Metal.

JAF-SE-81-107 Replace the existing unqualified NAMCO Snap-lock Position Switches which are designed for general purpose application only. The replacement NAMCO EA-740 and EA-180 Snap-Lock Limit Switches are qualified for harsh Environment use per DOR guidelines and NRC Bulletin 79-01B. New switches have been tested and qualified to the following IEEE standards: 323-1974, 344-1975 and 382-1972.

The new limit switch has the same mechanical and electrical characteristics as the one being replaced. The difference is in the higher quality materials used, which makes it suitable and qualified for harsh environments in Nuclear Power Plants. With qualified Conax N-11006-34 Seal Assembly, switch is also totally sealed from outside environment.

JAF-SE-81-108 This modification consists of adding a dual communication, adding an accumulator freeze card and adding a control card (Tap position control) to the Conitel Telemetering System which communicates information to ECC at Marcy. This equipment is electrically isolated from the power system and its sole function is to transmit information over a telephone line. It does not create a safety concern.

JAF-SE-81-109 The existing ASCO Solenoid Valves are designed for general purpose unqualified applications only. The replacement ASCO NP Series Solenoid Valves are qualified for harsh environment use per DOR guidelines and NRC Bulletin 79-01B. These replacement Solenoid Valves have been tested and qualified to the following IEEE standards: 323-1974, 344-1975 and 382-1972.

The replacement ASCO NP Series Solenoid Valves are physically interchangeable with the ones being replaced. The difference is that higher quality materials are used which makes them suitable and qualifiable for harsh environments in Nuclear Power Plants.

JAF-SE-82-005 Added a second scale for the following Reactor Water Level Instruments:

1) 02-3-LIS-101A thru 101D
2) 02-3-LIS-58A & B
3) 02-3-LIS-72A thru 72D
4) 02-3LIS-83A & B
5) 02-3-LITS-59A
6) 02-3-LITS-59B
7) 02-3-LI-85A
8) 02-3-LI-85B
9) 06-LI-94A thru 94C
10) 06-LR-97
11) 02-3-LI-86
12) 02-3-LIS-57A & B

With the addition of the second scale, Reactor Vessel Level instruments will be referenced to Top of Active Fuel. The dual indication will provide a direct indication of water level with respect to the fuel while still indicating the present levels. Since the present scales and setpoints (as required by Technical Specifications) will not be affected by this change, this does not constitute a safety question.

JAF-SE-82-006 This preoperational test was conducted to verify proper annunciation of the RPS protective circuitry as modified by Modification F1-81-10.

JAF-SE-82-009 This procedure covers the preoperational testing of the Suppression Pool Temperature monitoring system consisting of: sixteen (16) resistance temperature devices (RTD's) installed in the Torus shell one (1) Acurex Ten/5 Datalogger mounted in the 25-9 panel in the Control Room, and one (1) analog temperature indicator mounted on the 09-3 panel also in the Control Room. The test was essentially directed to:

1) Verify the integrity of the 16 RTD sensing loops and associated cable.
2) Verify the functionability of the datalogger, datalogger program, and the remote analog temperature indicator.
3) Verify the system alarms functionability.
4) Verify the calibration and accuracy of the system.

JAF-SE-82-010 This safety evaluation was for the test to verify proper HPCI Gland Seal Exhauster operation. By simulating an initiating signal and then removing it, the exhauster started and continued to operate until manually secured by the operator. The preoperational test was performed to verify proper operation of the HPCI Gland Seal Exhauster logic circuitry upon the completion of Modification F1-80-34 and to determine if an unreviewed safety question exists.

After a careful review it was determined that preoperational test 23A, does <u>not</u> in any way effect or alter the assumptions made in the FSAR or Technical Specifications and will <u>not</u> constitute an unreviewed safety question.

JAF-SE-82-013 This test was intended to verify that loss of RPS "B" power source, does not cause full isolation of the outboard MSIV's (29AOV-86A, B, C, D) and to verify outboard MSIV isolation logic after modification installation.

JAF-SE-82-016 This modification involved the removal of two vertical members interior to the frame of hanger 13-4A-NS-8. This hanger supports the RCIC pump discharge line and is the original Bergen-Paterson support H13-11.

During Inservice Inspection performed late in 1981, the above hanger was visually examined. The inspection revealed the apparent misalignment of the lugs and the frame. Subsequent exams showed that the pipe was off center from the frame, one lug caught against the edge of the frame itself.

Stress analysis performed by Stone & Webster showed that the hanger was not necessary for either operational support or seismic restraint. Therefore, the best corrective action for this problem was the removal of the vertical channel and plate where the lug was or could be caught, thereby preserving the vertical supporting characteristics of the hanger.

JAF-SE-82-022 The preoperational test for the Automatic Switchover of the RCIC Suction System encompassed the functional testing of the automatic actuation of the RCIC suction valves. Included was the functional testing of the associated annunciator windows.

This preoperational test was performed in order to verify the operability of the automatic switchover feature of the RCIC pump suction from the condensate storage tanks to the suppression pool upon reaching a low water level condition in the condensate storage tanks. This feature was installed as plant modification F1-81-05 in order to satisfy the requirements of NUREG-0737, Section II.K.3.22.

This preoperational test was performed during cold shutdown condition. Conduct of the test will not impact any statement in the FSAR or any Technical Specification.