James A. FitzPatrick
Nuclear Power Plant
P.O. Box 41
Lycoming, New York 13093
315 342-3840

New York Power Authority

Harry P. Salmon, Jr.
Resident Manager

August 12, 1992
JAFP-92-0598

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 1991

Enclosure: 1) Annual Summary of JAFNPP Changes, Tests, and Experiments for 1991

Gentlemen:

Enclosed is a summary of the changes, tests and experiments implemented at the James A. FitzPatrick Nuclear Power Plant during 1991.

This report provides the Nuclear Safety Evaluation number (e.g. JAF-SE-91-001) 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.

HPS:JFF:kls
ENCLOSURE

cc:

R. Liseno	D. Lindsey	M. Colomb
K. Vehstedt	D. Ruddy	A. Zaremba
R. Beedle - WPO	J. Gray - WPO w/encl.	RMS (WPO) w/encl.
JAFP File	RMS (JAF) w/encl.	

Cert # P 064 705 263

IE47

Introduction to the 1991 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:

- o 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;
- o Create the possibility of an accident of or malfunction of a different type than any previously evaluated in the FSAR;
- o 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: M1-86-078
JAF-SE-91-132 Core Spray Min Flow Valves Open Circuit Logic Change

This modification interlocks the valve 14MOV-5A/B auto open circuits with the respective Core Spray pump discharge pressure. A dedicated pressure switch to represent "pump running" was installed and wired in series with respective flow switches 14FIS-45A or 14FIS-45B. This logic change upgraded valve operation to meet containment isolation criteria.

This modification installed two pressure switches (14PS-42A,B), one on Rack 25-1 and the other on Rack 25-60. These switches were connected to the same sensing lines as the Core Spray discharge pressure switches for Automatic Depressurization system (ADS) Permissive (14PS-44A,B). The new switches were wired into the 14MOV-5A/B control circuits at junction box JB-R251A or JB-R2560A.

The installation of a pressure based permissive for the low flow auto-open logic was necessary to ensure that the low flow open signal did not maintain the Core spray minimum flow valves open when the respective Core Spray pump was not in operation. This modification allowed manual remote closure of the valve from the Control Room.

The purpose of this modification was to allow these valves to meet their remote manual containment isolation requirement by modifying the open circuit logic. The pressure switches have the same setpoint as the existing ADS logic pressure switches (14PS-44A/B). The setpoint is considered adequate for indication of pump running condition since it is used for the same purpose by the ADS. Pump protection is the primary safety function for the min-flow valves, therefore, keeping them normally open provides additional assurance of Core Spray pump availability. This modification allowed the valve to perform the containment isolation safety function by providing the capability to close the valve with the Core Spray pump not running.

The switch installations are QA Category I and Environmentally Qualified. They therefore meet the Safety Design Basis of the plant as stated in the FSAR. This design ensures continued reliable operation of the Core Spray and ADS systems and does not constitute any threat to safety systems since a failure of any one switch will not jeopardize both trains of Core Spray.

This modification did not change the closing time for the Minimum Flow Valves and therefore shall not change the assumption made in the 10CFR50 Appendix K 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: M1-87-087
JAF-SE-89-078 Change Setpoints of 70TS-109A, 70TS-109B, 70TS-110A, 70TS-110B

Temperature switches are provided to protect Control Room/Relay Room instrumentation in the condition of high ambient temperature by actuating the Control Room/Relay Room cooling system and initiating alarms. The modification lowers the setpoints of these switches from 104° to 98° F, reducing the probability of Control Room/Relay Room instrumentation malfunction due to high temperatures without affecting in any way other safety considerations associated with the switches, the Control Room/Relay Room cooling system, Control Room/Relay Room instrumentation, and related 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: M1-88-111
JAF-SE-90-075 Doorway Addition Between Work Control Center And Shift Supervisor Office

This modification provided direct passage between the Work Control Center and the Shift Supervisor's office by the addition of a doorway with sound curtain. This modification reduces unnecessary personnel traffic through the Main Control Room, and helps to maintain an efficient work environment for the Control Room general area. The new steel stud/sheetrock wall section and doorway could be readily installed in place of the existing blo wall since this structure is non-load bearing. The new wall and the emergency light are not within proximity to jeopardize any Control Room safety related equipment nor the Fire Protection Main Indicator Panel, however this modification still satisfies seismic class II design requirements in accordance with FSAR Section 12.4.6.3. Fire rating requirements are not applicable as this is part of an interior partition wall within the Control Room general 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: F1-88-116
JAF-SE-89-090 Addition of Valves, Nipples, and Caps to CRD Equipment Drains and Installation of Storage Cabinets for High Pressure Hoses

This modification provided for a temporary flowpath between the Control Rod Drive (CRD) withdraw vent valves and the Reactor Building Equipment Drainage System by utilizing a high pressure flexible hose. This action simplifies the performance of manually venting of individual control rods for alternate control rod insertion per Abnormal Operating Procedure AOP-34. There were no changes to the design function of the equipment drainage system or to the CRD HCUs.

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 M1-88-170
JAF-SE-90-097 Permanent Power to Rigging Test Stand and Tool Room Outlets

This Minor Modification Package authorized installation and documented update of the permanent power feed to the rigging test stand in the tool room of the Turbine Building Heater Bay. Also, the Minor Modification Package documented the power feed to the convenience outlets in the tool room.

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: M1-89-008
JAF-SE-90-106 Provide Permanent Air Supply to 33AOV-D-1

The installation of Temporary Modification 88-231 provided a temporary filter/regulator and lubricator for the air supply to 33AOV-D-1. This modification was necessary to document and provide an acceptable permanent air supply and control line to 33AOV-D-1. This modification documented the air supply to 33AOV-D-1 from outside the radiological barrier which allows access to the installed filter-regulator and lubricator. This 1/2" OD air supply line, along with the 1/4" copper air control line got an expansion loop to achieve flexibility for motion of the valve relative to the building structure. The two lines were permanently routed and supported. Air was routed to 33AOV-D-1 to permit rinse to hotwell per OP-[illegible].

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: M1-89-108
JAF-SE-90-109 Permanent Power to Weld Shop Compressor

This Minor Modification Package authorized installation and documented update of the permanent power feed to the weld shop compressor of the screenwell building.

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: M1-90-015
JAF-SE-91-029 Replacement of Three - 115KV North Bus Potential Transformers

This modification replaced the three - 115KV north bus potential transformers which did not consistently pass the Doble Test. The scope of this modification was to procure and install three - 115KV potential transformers (PTs) having the same technical characteristics as the existing potential transformers being replaced. The new PTs were installed in the same location as the existing PTs.

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. M1-90-031
JAF-SE-90-037 RWCU Pump Mechanical Seal Assembly Improvements for Performance

This modification improved the performance and reliability of the Reactor Water Clean Up (RWCU) pump mechanical seal. This modification modified the pump seal assembly and provides seal pump water pressure and flow rate indication. This modification removed the pumping ring from the mechanical seal and installed a blank flange to prevent back flow of seal water through the seal water heat exchanger. These changes were recommended by the seal manufacturer. Flow rate and pressure indication on the seal water supply was also added to allow monitoring by Operations of seal water flow parameters.

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: M1-90-198
JAF-SE-90-105 RES Work Area Expansion

This modification provided additional office space for Radiation Protection personnel near the existing Health Physics Office in the Administration Building, elevation 272'. The existing Whole Body Counter (WBC) room was converted into a Radiation Protection office by rearranging partition walls and relocating the WBC to a room on elevation 260' at the bottom of the adjacent stairwell.

No additional fire protection sprinklers or detectors were required to service these Radiological and Environmental Services (RES) work areas.

The relocation of the WBC was evaluated for a safe load transfer and rigging plan.

Two new 120 VAC electrical circuits were run to the new WBC Room to match the equipment's present electrical configuration. The total increase of electrical load on the existing non-safety lighting panel was small and was verified by a load analysis to be acceptable.

The existing air supply in the Locker Room had sufficient capacity to be extended to service the new RES office. The existing portable air conditioning unit at the bottom of the stairwell was adequate to accommodate the temperature and humidity-sensitive WBC equipment.

A radiological study was performed to ensure the suitability of the new location of the Whole Body Counter.

The relocation of the WBC to elevation 260' in the stairwell provides a minor increase in combustible loading in the area. Since there is no plant equipment in this area and because it is surrounded by 3-hour fire rated walls and doors, there was an insignificant effect on the Fire Protection 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.

[illegible] 1-90-227
[illegible] Rework of Main Generator Field Ground Alarm on
09-7-1 Annunciator Panel

An engineering evaluation determined that it was prudent to operate the plant with the main generator field ground detection circuit in the "Alarm only" mode. Consequently, annunciator window 09-7-1-28 was on continuously, reading "Main Gen Field Grd Will Alarm Only". The key couldn't be removed from switch 71-64F-1EXCN06, because it was in the left position. A protective tag was hung on the switch and the escutcheon plate has temporary labeling on it: left "Alarm and center "Trip".

In order to eliminate this condition, the following modification was done:

Switch 71-64F-1EXCN06 was rewired. In the left position, the generator automatically trips upon detection of a main generator field ground and the annunciator window 09-7-1-28 will come on. Annunciator window 09-7-1-28 was re-engraved to read "Main Gen Field Grd Will Trip". This is the "Trip" position of the switch with the key not removable in this position.

In the center position, switch 71-64F-1EXCN06 will be in "Alarm Only". This allows relay 71-64F1-1EXCN06 to activate annunciator window 09-7-1-27 "Main Gen Field Grd" upon the detection of a main generator field ground. The switch key is removable and the switch handle is in the vertical position. The switch escutcheon plate was replaced. This is the preferred operating position of the switch.

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: M1-90-250
JAF-SE-91-016 Condensate Storage Tank Vent Addition: 33TK-12A and 12B

This modification restored the condensate storage tanks (CST) inbreathing capacity to allow two 100% capacity core spray pumps to draw from the CSTs. This permits the 4665 GPM restriction imposed by OP-14 Section G to be removed. This envelopes the operation of HPCI and RCIC.

The addition of an 8" diameter vent ensures the internal vacuum pressure never exceeds the design pressure of 2" water column. To preclude any overflow to the ground, the new vent was extended to an elevation approximately equal to the existing 4" diameter vent. Wire mesh screen was installed over the open end of the vent for each CST. Existing heat tracing was extended from the drain line to the new vent.

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: M1-91-018
JAF-SE-91-014 Installation of Data Cable for the AVIION Computer System

This modification installed raceway and cable for the new Aviion computer which is located in the warehouse. This computer supports a network which is tied to other NYPA facilities.

This network was designed with flexibility and modularity so that NYPA will have the capability of growing as needs change. All new cabling to support the computer network was routed in existing telephone raceways. No plant tray systems were utilized. New raceway was installed to conform to plant standards where no telephone raceway was available.

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: M1-91-023
JAF-SE-91-112 Design Improvement for CRD Mechanism Seal, Bushing and O-Ring Spacer Plate

Modification M1-91-023 modified the CRD mechanism bushings and seals material from Graphitar 14 to Graphitar 3030. This change improved bushing and seal life. Additionally, the CRDM O-ring spacer design was slightly modified to improve ease of installation.

The new seals and bushings are manufactured from Graphitar 3030 (G-3030) versus the Graphitar 14 used in the old CRDM design. G-3030 is a high strength nickel-chrome impregnated graphite composite. There are no changes in the geometric configuration of the new parts and they are therefore interchangeable for use on existing CRDM's.

Two seals which use "C" springs were also additionally improved by modifying the forming and heat treatment of the springs. This reduced the installed springs stress by 35% and will make the springs less susceptible to IGSCC failure.

GE has performed extensive qualification testing of the improved materials and documented the results of this testing.

The O-ring spacer plate design was improved by adding a small lip (.005" X .010") on the top end of the three O-ring holes to capture the O-rings. This will alleviate difficulty in keeping the O-ring positioned properly during installation and reduce the potential of flange leakage. This design change will not affect spacer plate interchangeability between existing and new CRDM designs.

This modification required a revision to FSAR Section 5.3.3.1 which specifies CRDM bushing and seal material.

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: M1-91-088
JAF-SE-91-038 Modification to Suppression Pool Temperature Monitoring System with 15 RTDs Operable

This modification added a temperature bias in the software logic for the Suppression Pool Temperature Monitoring System to correct potential inaccuracies resulting from the following:

1. Utilizing only 15 out of 16 installed RTDs used to determine Suppression Pool Bulk Temperature

2. Differences between local and bulk temperature

3. Relative accuracy of the measurement system

The temperature bias ensures the JAF Suppression Pool Temperature Monitoring System will provide reasonable measure of bulk pool temperature in accordance with NUREG-0783 requirements.

The modified system applies the reading from one operable Resistive Temperature Detector (RTD) twice and averages it with the remaining 14 available RTDs to determine the bulk pool temperature.

This modification added a positive 4°F bias in the Suppression Pool Temperature Instrumentation Loop to provide conservative margins in the setpoints for alarm and operator actions required to prevent Technical Specification limits from being exceeded.

FSAR Sections 5.2, 7.3 and 14.6 were revised to reflect using 15 out of 16 RTDs and 4°F temperature bias added to the instrument loop.

General Electric evaluated the error in the bulk pool temperature measurement if heat is added non-uniformly to the pool with the system as designed and as installed. The evaluation was performed by using data on temperature distributions and profiles obtained from Monticello in-plant Safety Relief Valve tests.

The bulk pool temperature measurements are used to insure compliance with the FitzPatrick suppression pool temperature tech spec limits. These limits, which include an Limiting Condition for Operation (LCO) at 95°F, scram at 110°F and RPV depressurization at 120°F, are used in the containment pressure and temperature response analysis of FSAR Chapters 5.2 and 14.6 and the pool heat-up analysis of NEDC-24361-P.

Additionally, to meet the requirements of NUREG-0783, instrument loop accuracy and setpoints calculation was performed to ensure the relative accuracy of the measurement system is accounted for in the plant operating procedures and alarm setpoints.

The calculation result shows a positive 4°F bias needs to be added to the suppression pool temperature instrument loop to ensure the indicated temperature used by the plant operators to initiate manual actions does not result in violation of Technical Specification limits.

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: M1-91-124
JAF-SE-91-059 Emergency Service Water Keep Full

This modification provided a method for maintaining the Emergency Service Water (ESW) System full of water downstream of ESW pump discharge isolation valves 46MOV-101A(B) to the ESW check valves at the system interface with Normal Service Water (SWS) and Reactor Building Closed Loop Cooling Water (RBCLC). Damaging waterhammer events indicated that air pockets form in the idle ESW piping. The water in the ESW piping was draining back through 46MOV-101A and -101B when the ESW pumps were idle. The service water connecting lines will now maintain water pressure on the ESW piping.

This modification consisted of adding a line from service water supply header 46-6"-WS-151-37, connecting into both "A" and "B" ESW lines. A check valve was installed in each new keep-full line for safety-related system boundary isolation. Certain ESW high points also had vent lines with manual valves added to remove air from the ESW System after system maintenance, if required, and to help diagnose when and how air enters the 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: F1-91-138
JAF-SE-91-098 Installation of Hot Water Boiler 87HWB-1B

This modification installed a hot water boiler to supply hot water to the non-safety related plant heating system, while auxiliary boiler 87AB-1A and hot water heat exchangers 87E-24A and -24B are inoperable.

This modification installed a packaged boiler, associated piping and required electrical supply to the hot water boiler. This hot water boiler was located in space vacated by removal of Auxiliary Boiler 87AB-1B.

This modification also restored piping and accessories, conduits, cables, lighting fixtures, platforms, wall siding, beam and doors, which were removed to facilitate removal of the Auxiliary Boiler 87AB-1B under Modification No. F1-91-178.

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: F1-91-138 Test: POT-87A
JAF-SE-91-110 Installation of Hot Water Boiler

This pre-operational test demonstrated that the pre-operational testing of the hot water boiler 87HWB-1B and its associated controls and components did not adversely affect the safety of the plant.

This test could be performed when the plant is operating or shutdown.

Components verified included the boiler fuel oil pump, combustion air blower, atomizing air compressor, new supply and return hot water lines, fuel oil supply and return lines.

Due to the amount of modification work performed in the Auxiliary Boiler Room, the pre-operational test procedure functionally tested existing hot water and glycol circulation loops prior to operation of the boiler.

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: M1-91-156
JAF-SE-91-061 Replacement of Control Switches for 10MOV-12A/B; Justification for LPCI Throttling with 10MOV-27A/B and Shutdown Cooling Throttling with 10MOV-12A/B and 10MOV-66A/B

This modification ensured long term reliability of the Residual Heat Removal (RHR) system by reducing the overall service of the outboard Low Pressure Coolant Injection (LPCI) valve as a throttling device. Since the majority of the throttling service is provided in the shutdown cooling mode of RHR, this modification moved the shutdown cooling throttle function to the RHR pump discharge restriction orifices.

This modification replaced the existing keylock open/shut Control Room switch on the 09-3 panel with a three position open/neutral/shut switch for the RHR heat exchanger outlet valve (10MOV-12A/B). This allows remote throttling of 10MOV-12A/B from the Control Room. Following implementation of this modification, the RHR system will be operated at maximum flow rate and cooldown rate will be maintained by throttling either 10MOV-12A/B or 10MOV-66A/B. The outboard LPCI injection valve will only be throttled to control RHR system flow rate during post accident LPCI conditions.

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: M1-91-156 Test: POT-10P
JAF-SE-91-078 RHR Shutdown Cooling Valve Throttling

This pre-operational test verified that 10MOV-66A/B and 10MOV-12A/B are capable of controlling flow through the RHR heat exchanger during shutdown cooling. This test simulated conditions that 10MOV-66A/B and 10MOV-12A/B will normally experience during RHR shutdown cooling operations. Pertinent valve and pump vibration, pump current demand, and pump winding temperature data were obtained. In addition, this pre-operational test ensured that the RHR pump is capable of providing flow near runout conditions for extended periods of time without damaging the pump or motor.

During the performance of this test, the reactor was maintained in a cold 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 Specifications or an unreviewed safety question as defined in 10CFR50.59.

Modification: M1-91-168
JAF-SE-91-064 Building & Grounds New Office Fire Protection Modification

This modification extended and modified the existing sprinkler layout to provide fire protection to newly constructed office areas in the new Buildings & Grounds Office Building (a former Warehouse).

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: M1-91-170
JAF-SE-91-068 "B" SRV Tailpipe Drain Line

A gross snubber failure had occurred and was discovered on the "B" SRV tailpipe during 1990 Refuel Outage snubber inspections. This line was inspected during the 1991 Maintenance Outage and was found to have reverse slope at the beginning of a long horizontal run with 90 elbows and the failed snubber located approximately 80 feet downstream. This configuration, combined with slow leakage past the SRV, allowed for a waterhammer to occur during steam blowdown. Additionally, a root cause evaluation at the failed snubber and a review of the pipe stress analysis provided conclusive evidence that the snubber broke due to a steam discharge with a water slug during test actuations in either 1987 or 1988.

This modification provided a small bore drain pipe at the low point on line 10" SSV-302-1B to prevent condensate from accumulating, and thereby preclude the potential for the reoccurrence of a waterhammer. With the implementation of this modification, reworking the "B" SRV tailpipe and supports to change the slope was not required, and Action Record #1769 was resolved.

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: M1-91-172
JAF-SE-91-066 Control and Relay Room Cabinet Anchorage Verification and Modification

Existing Control Room cabinets were not adequately anchored to the concrete floor. This modification provided additional welding to restore these cabinets to meet the original design basis requirements.

The balance of all the panels in the Main Control Room and Relay Room were inspected, and the anchorage configuration of the cabinets was evaluated and documented.

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-[illegible]-177
JAF-SE-91-069 1 Hour Fire Wrap of Conduit 1CK205RB

Electrical cables associated with fans 73FN-3A and 73FN-3B (ventilation to the Emergency Service Water Pump Rooms) were routed through the same fire area. Both ventilation fans are identified as Safe Shutdown components. The two redundant cables were routed with intervening combustibles (electrical cables) present.

To meet 10CFR50, Appendix R, Section III.G.2 separation and fire protection requirements this modification consisted of the installation of a one hour fire rated HEMYC ceramic fire blanket system on conduit 1CK205RB and associated supports and 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: F1-91-178
JAF-SE-91-093 Removal of Auxiliary Boiler 87AB-1B

This safety evaluation addressed the demolition and removal of Auxiliary Boiler 87AB-1B from the inoperable Auxiliary Boiler System. Removal of Auxiliary Boiler 87AB-1B was required after it was radioactively contaminated.

No safety related equipment is located in the Auxiliary Boiler Building. Fire Water line 76-12"-WF-151-8 which penetrates the auxiliary boiler room floor near Col/Line 7G was within approximately 2 feet of the removal path of the boilers. In the unlikely event the boiler sheared this pipe, the line would have been isolated by closing 76PIV-22 and 76FPS-109. This would have resulted in 20 manual fire hose stations located in the Auxiliary Boiler building, Administration Building and Reactor Building becoming inoperable. Six of these twenty hose stations would have required back-up protection in accordance with Technical Specification Section 3.12D.

The portions of piping which were removed or abandoned have no impact on plant safety. Each system which enters the Auxiliary Boiler Building was isolated from the remaining plant by manual isolation valves.

All piping which may have been contaminated or abandoned in place and not returned to service when the hot water boiler was installed was capped.

Cutting service water line 45-1"-WS-151-108E, isolating portions of service air connections, cutting Turbine Building Closed Loop Cooling Water Supply and Return lines, and cutting the instrument air lines which cross the east wall had no impact on plant safety.

Removal of cables in the removal path did not impact the Appendix R safe shutdown systems, equipment or access as determined by a Fire Protection/Appendix R compliance review performed in accordance with NYPA procedure IES 4.2.

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: F1-91-178
JAF-SE-91-095 Substitution of the Auxiliary Boiler Steam Supply

During power operation, heating steam is normally provided by the Reboiler system steam to High Temperature Hot Water Heat Exchangers 87E-24A, -24B to heat the plant's hot water heating system. This forced circulation hot water heating system provides building recirculation air heating and heats the hot water ethyleneglycol system for heating outside air drawn into plant ventilation systems.

This modification installed a substitute hot water boiler which provides heat for plant's hot water heating previously provided by steam to the High Temperature Hot Water Heat Exchangers. This new boiler ties directly into the non-contaminated hot water heating portion of the Auxiliary Boiler system. Heat Exchangers 87E-24A & 24B will be isolated when the Reboiler system is not in service.

A study of the Auxiliary Boiler system actual heat load found that the new boiler has sufficient capacity to meet the actual plant heat load.

The hot water boiler will be fueled by No. 2 fuel oil, which will be supplied by an oil tank truck located in the diked oil unloading area adjacent to the above ground fuel oil tank (87-TK-49). Adequate environmental protection measures have been and will be taken to contain possible oil leakage or spills. The New York State Department of Environmental Conservation (DEC) has granted permission to operate the hot water boiler without applying for a change to the Air Quality Permit to burn No. 2 fuel.

In the event temperatures in buildings containing other safety related equipment cannot be maintained above the environmental qualification lower limit temperature of 40°F, the plant will be shut down.

The Auxiliary Boiler system is described in Section 9.9.3.2. This section will be revised to describe the substitute sources of heating which will replace the steam previously provided by the Auxiliary Boiler system. Use of Auxiliary Boiler steam for turbine gland sealing during plant start-up and shutdown is also described in FSAR Section 10.4.3.2. However, auxiliary boiler steam is not used for gland sealing because it would result in mixing the auxiliary boiler water chemistry into the condensate system. This section will be revised to reflect the current method of turbine gland sealing using main steam and the reboiler, and to delete reference to the auxiliary boiler.

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: M1-91-196
JAF-SE-91-072 Isolation of Crescent Area Unit Coolers for Appendix R Fires

Control and indications for the unit coolers are on panel 09-75 in the Main Control Room. In the event of a fire in the Main Control Room, Relay Room or Cable Spreading Room, loss of control power to the unit coolers (due to a short circuit or hot short in the unit cooler circuitry) could occur. The circuitry was modified such that, if a fire in the Main Control Room, Relay Room or Cable Spreading Room occurs, Div. II crescent area unit coolers can be isolated and will be operable from control panel 66HV-3B located in the Reactor Building.

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: M1-91-196 Test: POT-66H
JAF-SE-91-077 Pre-operational Test Procedure for Isolation of Crescent Area Unit Coolers for Appendix R Fires

This pre-operational test verified the proper operation of the equipment installed under Modification M1-91-196, Isolation of Crescent Area Unit Coolers for Appendix R Fires. The control circuitry for Div. II unit coolers including the installation of isolation switches, the new control fuses and the wiring changes were checked.

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

Modification: M1-91-210
JAF-SE-91-080 10MOV-25A & B, 14MOV-12A & B Bonnet External Bypass Line Installation

This modification provided for the installation of a small bore bypass line on 10MOV-25A & B from the bonnet area of the valve to piping located on the reactor side of the valve. This bypass line provides a relief path for pressure equalization between the valve bonnet area and the reactor side piping. The new bypass line can be isolated from the reactor side piping by closing 10RHR-786A & B.

This modification also installed a small bore bypass line, including a manual isolation valve, on each of the Core Spray inboard isolation valves 14MOV-12A & B. These lines run from the bonnet area of the valves to the reactor side of the valve body. These lines were routed to avoid the valve crotch and guide areas.

The bypass piping installed on valves 10MOV-25A & B and 14MOV-12A & B provided an effective solution to the "pressure cocking" phenomenon associated with these 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: M1-91-219
JAF-SE-91-088 Relocation of Fire Hose Station No. 142

Fire Hose Station No. 142 was attached to the east wall of the Auxiliary Boiler Building. The east wall, between column lines 6 and 7, was removed to facilitate removal of Auxiliary Boiler "B" and installation of a new hot water boiler.

Isolation of this hose station required isolation of twenty hose stations in the Administration Building Fire Loop. Six of the twenty Administration hose stations are required to be operable and their isolation required back-up protection/coverage. To minimize the duration of backup fire protection required and the work necessary for temporary removal and reinstallation, Fire Hose Station No. 142 was relocated such that it did not interface with the path for removal/installation of the Auxiliary Boilers.

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: M1-91-251
JAF-SE-91-101 HPCI Exhaust Check Valve 23HPI-65 Test Connection Line Modification

This modification reduced the mass of the branch piping by permanently removing a portion of the piping and components. The reduced mass greatly improves the ability of the piping to resist fatigue failure. HPCI system design and operation was not adversely affected by this modification since it only involved changes to the piping used for leak rate testing. Surface examination was performed on the new welds in accordance with the requirements of the FSAR. Inservice leak testing was performed in accordance with the requirements of the original construction code.

The High Pressure Coolant Injection System is defined as a QA Category I safety-related system per FSAR Section 12.2. This modification affected a small bore pipe that was used for leak rate testing of primary containment isolation valves.

This modification was performed on a portion of piping that forms a primary containment pressure boundary. JAF Technical Specification paragraph 4.7.A.2.f requires that a leakage rate test be performed to confirm the containment integrity of the area affected by the modification. Local leak rate testing was performed to confirm the piping and containment integrity in this location.

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: Temporary Modification 91-010
JAF-SE-91-005 Temporary Electrical Monitoring of B Reactor Water Recirculation M-G Set

This Temporary Modification 91-010 provided monitoring of electrical parameters of the "B" RWR M-G Set to determine the cause of recent trips of the machine. The parameters monitored were associated with the voltage regulation of the generator and included: The tachometer generator output voltage, the exciter field voltage, the generator field voltage, and the potential across the voltage adjust rheostat.

Troubleshooting required operating the machine at or near rated speed and voltage.

The monitoring was done using high impedance test instruments such as a strip chart recorder, an oscilloscope and a multimeter. The test connections were made at terminal strips or circuit board pins in the M-G Set relay panel.

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: Temporary Modification 91-037
JAF-SE-91-022 Reactor Building Perimeter Drain Temporary Discharge Drain Path

This temporary modification segregated mat drainage for the Reactor Building from west storm drain system to reduce water processing requirements. This work involved disconnecting the discharge line for 75P-4A, sampling installing temporary flexible hose and routing the discharge south to discharge into the site south drainage ditch.

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: Temporary Modification 91-038
JAF-SE-91-021 Safety Evaluation for Temporary Diversion of Storm Drain Through a Filter Unit to Circulating Water Discharge Canal

This temporary modification installed seals at the outfall of the west storm drain to block the flow of the storm sewers to the lake. This modification also installed a temporary filtration system to process this water currently being collected in the west storm drain.

This safety evaluation determined the acceptability of processing the potentially radioactively contaminated water in the west storm drain. The water collected in the west storm drain was potentially contaminated due to a release of radioactive materials at the JAF Nuclear Power Plant on March 18, 1991.

The west storm drain was pumped into temporary containers. This water was then pumped through a temporary demineralizer system consisting of a series of demineralizer towers and a charcoal filter. This water was then pumped to storage containers for sampling. If the sampled water was within plant discharge limits, the water was pumped to the circulating water discharge canal. Drain water not meeting plant discharge quality limits, was temporarily stored in the storage containers for future processing.

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: Temporary Modification 91-039
JAF-SE-91-020 Administration Building Roof Drain Temporary Discharge Drain Path

This modification segregated potentially contaminated Administration Building roof water drainage from the west storm drain system. A 6" PVC piping system was connected to existing roof drain lines in the Administration Building Fan Room and routed to a collection container at ground elevation located on the west side of the Administration Building.

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: Temporary Modification 91-040
JAF-SE-91-023 Installation of a Temporary Filtration System in the Liquid Radwaste System

This temporary modification provided an alternate means of filtering radioactive waste that would normally be processed by the waste concentrators (20EV-648A/B). The temporary filtration system was connected by means of high pressure hoses to the concentrator feed pump discharge line common header and to the Radwaste Precoat Tank (20TK-23) overflow line. The existing overflow line directed liquid radwaste from the precoat tank to either the Waste Collector Tank (20TK-11) or to the Waste Neutralizer Tank (20TK-2642A/B). The capacity of the temporary system exceeded the filtering capacity of the existing concentrator.

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: Temporary Modification 91-060
JAF-SE-91-026 Temporary Installation of Steam Boiler to Provide Steam to the Nitrogen Vaporizer

This temporary modification provided a temporary boiler to supply steam to the non-safety related nitrogen purge steam vaporizer, while the auxiliary boilers were inoperable. Steam to the nitrogen vaporizer was only required for containment inerting in support of plant startup.

The temporary modification provided a trailer-mounted boiler, associated piping and required electrical supply in the area of the nitrogen storage enclosure.

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: Temporary Modification 91-083
JAF-SE-91-037 Acoustic Fish Deterrence Project

This modification involved the temporary installation and operation of acoustical devices on and near the Circulating Water Intake to deter fish (alewives) from entering the intake and impinging in the screenwell area. The acoustic components were installed remote from the nearest safety related equipment, components or structures. The mounting of the devices was redundant to prevent loss of items into the intake.

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: Temporary Modification 91-092
JAF-SE-91-039 Installation of Temporary Demineralizer Resin Beds in the Make-up Water Treatment System

The anion, cation, mixed bed portion of the make-up water treatment system was inoperable due to undetermined blockage within the mixed bed vessel.

This modification provided an alternate means of processing demineralized water to be used in the make-up water treatment system. A temporary demineralizer system replaced the existing Cation, Anion, and Mixed Bed units. The supply to the temporary demineralizer was connected to the discharge of the activated carbon filter by means of flexible hoses. A flexible hose also directed the effluent of the temporary unit to either the demineralized water storage tanks or to an existing reverse osmosis unit.

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: Temporary Modification 91-224
JAF-SE-91-114 Chemical Cleaning of the Service Water System

Small bore piping in the Service Water System (SWS) and Emergency Service Water (ESW) System has corrosion in the form of nodules which results in reductions of the piping cross-sectional area. This reduced the flow of water to system equipment being supplied. To correct this problem, on-line chemical cleaning of the service water system will be performed. Various chemicals will be injected at the normal service water pump forebays, taken into the system by the SWS pumps, pass through the service water system, and then discharged to the lake. These chemicals are absorbed by the corrosion nodules which are then dissolved and dispersed into the service water flow and then to the lake. This process can be performed during all plant operating modes.

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

Modification: N/A
JAF-SE-88-019 Pressure Seal Restoration of 13AOV-22

The surface of the pressure seal area of the Reactor Core Isolation Cooling (RCIC) testable check valve (13AOV-22) was restored after drilling two holes through the valve neck which were necessary to perform an operational leak repair. The pressure seal area was restored by drilling tapered holes, slightly larger than the leak repair holes, and installing tapered pins to support a ½" deep carbon steel, structural weld on the outside surface and a austenitic stainless steel weld clad on the inside surface. The weld clad on the inside surface was finished smooth by honing.

The valve pressure seal failed, during normal operation, causing an excessive amount of reactor coolant to flash through the bonnet of the valve. A permanent repair was not practical while operating since it would be necessary to isolate reactor feedwater in order to isolate 13AOV-22. Therefore, 13AOV-22 was leak repaired by drilling holes through the pressure seal area and injecting leak repair sealant while operating.

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

Modification: N/A
JAF-SE-90-042 General Heavy Load Handling System Requirements to Meet NUREG-0612 Criteria

Heavy loads such as new plant equipment and containers (defined as greater than 750lbs. at FitzPatrick) are frequently lifted or moved around the plant. A generic evaluation which addresses the requirements for rigging and safe load paths consistent with NUREG-0612 philosophy for each lift will expedite heavy load movements and reduce the need for an individual safety evaluation or engineering assessment of each heavy load lift.

All manufactured components including slings, hoists, shackles, links, Hilti-Kwik bolts, and like components, lifting lugs, and cask trunions of the haaandling system shall have a minimum design safety factor of 10:1 (ultimate strength to the actual lift load). All structural steel components including plates, bars,and structural shapes shall have a 5:1 minimum design safety factor with respect to the ultimate strength of the material for the maximum combined static and dynamic load, and also a 3:1 minimum design safety factor with respect to the yield strength of the material for the maximum combined static and dynamic load. These actions ensured that the requirements for single-failure-proof lifting systems as detailed in NUREG-0612 will be satisfactorily addressed.

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

Modification: N/A
JAF-SE-90-082 Removal and Reinstallation of the Fuel Preparation Machine From/Into the Spent Fuel Pool

The north fuel preparation machine, one of two machines located on the east wall of the spent fuel pool, had been out of service for an indefinite period of time. The evaluation allowed the assembly to be removed from the spent fuel pool, repaired, and reinstalled or raised approximately 5 feet and rotated 180°.

The weight of the fuel prep machine is approximately 775 lbs. and is, therefore, classified as a heavy load per NRC NUREG-0612.

A specific plant procedure was issued to control the activities related to the rigging arrangement, safe load path, radiological monitoring, cleaning requirements, and the temporary storage/enclosure requirements.

The assembly was supported from two 1" lifting eyes bolted to the fuel prep machine platform. The lifting eyes, slings, and other miscellaneous rigging equipment had a minimum 10:1 safety factor to material ultimate strength.

The reactor building crane 20 ton auxiliary hook carried the fuel prep machine.

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

Modification: N/A
JAF-SE-90-102 Analysis for Feedwater Pump Discharge Check Valves (34FWS-4A&B) Lost Parts

The purpose of this evaluation was to demonstrate that safe reactor operation will not be compromised with the feedwater pump discharge check valves (34FWS-4 A&B) lost parts remaining inside the reactor vessel. The evaluation addressed the safety concerns of fuel bundle flow blockage, control rod operation interference and corrosion or chemical reactions with other reactor materials.

The lost parts from the feedwater A pump discharge check valve (34FWS-4A) are two anti-rotational lugs and two dowel pins. Parts of a cotter pin are missing from the B pump discharge check valve.

The evaluation concluded:

> The small size of the parts along with the high flow velocity in the pipe, indicated that there is no likelihood for any of the lost parts to be caught at a location along the flow path or be trapped inside a valve and prevent its normal operation.
>
> An object entering the lower plenum via an operating jet pump will be driven inward along the vessel bottom and will most likely come to rest near a sheltered location among the stub tubes.
>
> If trapped inside the cavity, the parts most likely will be trapped against the lower tie plate grid where they will cause partial blockage of the bundle flow. However, the resulting bundle flow reduction is estimated to be about 15%. This degree of blockage is less than the 86% flow blockage criteria required to initiate boiling transition in a bundle.
>
> If trapped in a fuel bundle, the likelihood for any fuel clad fretting wear due to the dowel pins was deemed negligible because of the random motion of the parts inside the fuel bundle.

During a refueling outage, as the fuel bundle is being moved, there is a remote possibility that the parts could be shaken loose from the bundle and fall into a control rod guide tube opening. The anti-rotational lugs would come to rest on top of the velocity limiter, but that would not interfere with control rod insertion.

There is no potential for interference with control rod operation.

Since the lost parts are made from stainless steel, there is no potential for corrosion or chemical reaction with other reactor materials.

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

Modification: N/A Test: POT-01-107F
JAF-SE-91-018 Safety Evaluation for Special Test for Offgas System Leak Testing to Assess Pressure Integrity

This test assessed the potential system damage resulting from an apparent hydrogen ignition in the Offgas System on March 18, 1991. This test evaluated the system integrity to provide sufficient environmental and radiological controls to ensure the safe performance of this procedure.

This leak test involved pressurizing those portions of the system which may have experienced a pressure transient to verify system integrity. The plant compressed air system was used to pressurize this piping. The test rig consisted of a check valve to protect the compressed air system from contamination, air filters and a pressure regulator, a fill valve, an outlet valve, a calibrated relief valve set at 20 psig and a calibrated test pressure gage.

The Offgas System areas tested were within the normal radiological restricted areas or buried underground. The accessible restricted areas and the stack were continuously monitored for airborne activity to minimize any potential release. The stack effluent radiation monitors were monitored continuously during the test to minimize any potential uncontrolled release past the stack isolation valve from exceeding the Technical Specification limits.

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: N/A
JAF-SE-91-019 Change to FSAR Section 11.4 Concerning Offgas Filter Testing Requirements

This Nuclear Safety Evaluation presented analysis of a change to Section 11.4.8 of the Final Safety Analysis Report (FSAR) which describes the inspection and testing of components in the Radioactive Gaseous Waste System (01-107). This section was mis-labeled as 11.4.7. Specifically, this evaluation supported changing the discussion about testing the offgas filters 01-107F-1A/B to correct a technical error and reflect actual operating practices.

The current FSAR testing description of offgas filter testing was inaccurate and if attempted would be extremely dangerous. Ensuring proper filter performance is adequately accomplished by correctly installing certified filters and DOP testing prior to initial service. The radioactive gaseous effluent monitoring program verifies that doses to the public for gaseous radioactive effluent is maintained ALARA.

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

Modification: N/A
JAF-SE-91-028 Evaluation of the Circulating Water System Reverse Flow Gate Blockage Due to Sand Accumulation

During the 1991 Maintenance Outage divers inspected the Circulating Water System reverse flow tunnel and gate. Inspection revealed it to be virtually full of sand and silt for approximately two-thirds the length of the tunnel (20 feet) to within 3" from the top of the tunnel. This condition potentially prevented the plant from using the reverse flow tunnel if it were required.

Flow velocity was high enough to clear a significant opening in the tunnel. The flow velocity would then decrease in the intake structure and the silt and sand would drop out. This intake area is well removed from the pumphouse forebay and sand would not have been drawn into the emergency pumps.

Plant operation with the sand accumulation in the Circulating Water System reverse flow tunnel did not impact plant safety. At that time of year, (late winter) Lake Ontario was starting to warm and virtually all ice on the lake had melted. Therefore, the reverse flow capability would not have been required until the winter of 1991-1992. If the reverse flow function were required in the past, it would have performed its function. The reverse flow tunnel was cleaned during the 1991/1992 Refuel Outage to ensure the reverse flow capability will be available during the winter of 1991-1992.

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

Modification: N/A
JAF-SE-91-034 East Cable Tunnel Sump Pump Isolation for 75P-1A, 1B, 3A & 3B

This temporary isolation of the east cable tunnel sump pumps reduced the potential for an unmonitored radiological release to the environment. A condensate return line (20-4"-C-136-35) is located in both the East and West Cable Tunnels. The floor drains in the tunnels are routed to either the north or south sumps located in the East Cable tunnel. The south sump is pumped out to the West Storm Drain System via Manhole No. 2. The north sump is pumped to the ESW/Fire Pump/Service Water Forebays. Therefore, if the condensate line failed, the potential existed for an unmonitored release to the environment via the storm drains or the ESW/Fire Pump/Service Water Forebays.

This temporary isolation shall remain in effect until such time that a permanent solution can be addressed.

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-91-040 Radioactive Material Storage On-site (Valve Shed)

This evaluation was to allow the "Valve Shed", located inside the protected area, to be utilized as both a staging area for low level radioactive materials and as a warehouse for storing non-contaminated oils and resins.

The arrangement for low-level radioactive material storage was inefficient and scattered throughout various areas of the restricted area of the plant. The use of the "Valve Shed" provides controlled temporary storage of the staging material, maintenance tools, and equipment that have low-level radioactive contamination. Since the movement of the material, the old "Fab Shop" area, where some low level radioactive materials were being stored, has been renovated and released as a clean area.

The staging area doors of the "Valve Shed" shall be kept locked to restrict access and control contamination. Loose surface contamination levels of the stored material will be limited to <1000 dpm/100cm^2.

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

Modification: N/A
JAF-SE-91-044 Radwaste Enhanced Settling Program

The purpose of this program was to determine if the addition of organic polyelectrolytes into a Radwaste System Waste Neutralization Tank (WNT) could reduce the amount of suspended solids in the WNT, thereby reducing the solids loading on downstream process equipment. An added benefit was that the agglomerates of solids formed are less dense than packed solids and thus would be easier to transfer during WNT desludging operations.

This program was conducted by adding a small amount of polymer to a WNT via the waste collector filter precoat tank. Sampling and analysis of the WNT contents before and after the addition of polymer was performed in order to determine the effectiveness of the program. If the program is successful, it is planned to add polymer to the WNT on an "as needed" basis, depending upon the suspended solids content.

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

Modification: N/A Test: POT-10M
JAF-SE-91-065 RHR Shutdown Cooling Throttling Valve

This pre-operational test verified that 10MOV-66A/B and 10MOV-12A/B were capable of controlling Residual Heat Removal (RHR) shutdown cooling system flow rate from the Control Room. In addition, this pre-operational test ensured that the 10MOV-12A/B valves were capable of opening against a differential pressure equal to RHR pump head at minimum flow conditions. This test simulated conditions 10MOV-66A/B and 10MOV-12A/B will normally experience during RHR shutdown cooling operations and obtained vibration data.

During the performance of this test, the reactor was maintained in a cold condition. While an RHR pump was recirculating water from torus to torus, the opposite RHR/LPCI subsystem was available for shutdown cooling.

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: N/A Test: POT-10N
JAF-SE-91-073 Troubleshooting of 10MOV-25B Motor Failure

This test was to troubleshoot the failure of the motor on 10MOV-25B. Testing performed prior to the motor failure was repeated to evaluate whether pressure induced locking of the valve occurred.

During the performance of this test, the reactor was maintained in a cold condition. While the "B" RHR subsystem was inoperable for testing, the "A" RHR/LPCI subsystems was available for shutdown cooling or emergency use.

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: N/A
JAF-SE-91-076 Evaluation Of The Post-Accident Hydrogen Monitoring System

This Nuclear Safety Evaluation determined that there were no safety concerns associated with continued operation of JAFNPP without a fully redundant Post Accident Hydrogen Monitoring System. The "B" train of the Post Accident Hydrogen Monitoring System did not have the capability to obtain a gaseous sample of the torus atmosphere.

Through existing equipment Hydrogen and Oxygen concentration was continuously monitored to remain in compliance with JAF Technical Specifications. Alternate equipment or methods were available to continue to monitor H2/O2 concentration if the existing Post Accident Hydrogen Monitoring System became inoperable. Finally, even with a complete loss of H2/O2 concentration monitoring capability, the control room operator had alternate methods in which to safely mitigate this condition.

NYPA has also committed to install a separate redundant sample line to monitor the torus free atmosphere. Modification F1-91-032 will install the redundant torus sample line during the 1992 Refueling Outage.

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

Modification: N/A
JAF-SE-91-081 Operability Justification For SRV Acoustic Monitoring System Passive Channels G-L With Increased Background Noise

This Nuclear Safety Evaluation demonstrated that the Safety Relief Valve (SRV) Acoustic Monitoring System can fulfill its design intent and be declared operable with 5 of 11 passive (backup) channels detecting higher than normal unfiltered background noise levels.

The design of the SRV Acoustic Monitoring System is such that the existence of higher than average background noise out of the band of interest will not impair the ability of the acoustic monitors to detect SRV position and alarm as designed. Therefore, plant operation was determined acceptable with G-L passive monitors receiving higher than normal, unfiltered 60 Hz background noise.

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

Modification: N/A
JAF-SE-91-118 Temporary Storage of the Radioactively Contaminated Auxiliary Boiler on Site

During an unplanned evolution in March, 1991 the operating Auxiliary Boiler became radioactively contaminated. Subsequently, the auxiliary boiler system was determined inoperable and it was decided to remove the boiler.

The boiler was then stored directly outside the Auxiliary Boiler Building. Provisions have been made for ultimate disposal.

The boiler internals have been dried and sealed by welding. All manways, water and air lines, and other orifices were welded shut.

The radioactive material stored within the auxiliary boiler was located within the protected area. The radioactive material contained within the boiler was sealed, and thus, not readily transported to the environment.

Radiation exposure to the public and workers was controlled by posting the area and by situating the material within the protected area of the site. The radiation level at the boundary of the auxiliary boiler location was less than or equal to 0.4 MR/hr and loose contamination was less than one thousand (1000) dpm/100 cm2.

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

Modification: N/A
JAF-SE-91-121 Temporary Storage of Radioactively Contaminated Soil on Site

During an unplanned evolution in March, 1991 approximately twenty to thirty thousand cubic feet (20,000 - 30,000 ft3) of soil became radioactively contaminated. Subsequently, the soil was removed from specific areas around the plant and loaded in a pile on the east side of the FitzPatrick site. The soil was sealed within a tarpaulin and occupied an area of approximately 45 feet by 150 feet.

The radioactive material stored within the tarpaulin structure was located within the protected area. Radiation exposure to the public and workers was controlled by posting the area and by situating the material within the protected area of the site. The radiation level at the boundary of the soil was less than or equal to 0.4 MR/hr.

The soil was expected to remain where it was located, until shipping and disposal was arranged.

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

Modification: N/A
JAF-SE-91-135 Evaluation of ESW and RHRSW Systems Operability due to EDG Jacket Water Cooler Return Piping to the Emergency Pump Bay

The Design Basis Document reconstitution effort identified a design deficiency in the Emergency Service Water (ESW) System. The ESW return piping from the four Emergency Diesel Generator (EDG) jacket coolers is directed back into the A ESW pump bay. This could result in an inlet temperature increase at the ESW and Residual Heat Removal Service Water (RHRSW) pump suctions due to this recirculation flow.

The analyses performed by SWEC are conservative and might appear to conflict with previous plant experience wherein the EDG's were run without (an) RHRSW pump(s) in operation and high jacket water temperature alarms were not received. The long-term resolution of this issue to permit plant operation at lake water temperatures above 65°F will require either a physical modification to reroute the jacket water return line or extensive analysis and testing of the existing configuration with subsequent comparison to the analytical models used by SWEC. Until either of these actions is performed, the effect of EDG jacket cooler recirculation on the ESW and RHRSW pump inlet temperature limits the maximum allowable lake temperature for safe operation of the plant to 65°F with provisions to start RHRSW pumps within 30 minutes of an EDG start. This prevents exceeding the maximum analyzed ESW and RHRSW supply temperature limit of 82°F.

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