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Nuclear Power Plant
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New York Power Authority

Michael J. Colomb
Plant Manager

April 19, 1996
JAFP-96-0168

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, D.C. 20555

Subject: **James A. FitzPatrick Nuclear Power Plant**
Docket No. 50-333
Annual Summary of Plant Changes, Tests, and Experiments for 1995 as Required by 10 CFR 50.59

Attachment 1: Annual Summary of Changes, Tests, and Experiments for 1995

Dear Sir:

Attached is a summary of the changes, tests, and experiments implemented at the James A. FitzPatrick Nuclear Power Plant during 1995.

This report provides the Nuclear Safety Evaluation number (e.g. JAF-SE-95-0001), revision number, title, modification number, if applicable, followed by a brief description of the corresponding change, test, or experiment and safety evaluation summary as required by 10 CFR 50.59(b)(2).

If you have any questions concerning this report, please contact Mr. A. Zaremba of my staff at (315) 349-6365.

Very truly yours,

MICHAEL J. COLOMB

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ATTACHMENT 1 TO JAFP-96-0168

<u>Annual Summary of Changes, Tests, and Experiments for 1995</u>

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* These Nuclear Safety Evaluations were incorporated prior to 1995. Audit of records determined that these items were omitted from previous 10 CFR 50.59 submittals.

Introduction to the 1995 Annual 10 CFR 50.59 Report

10 CFR 50.59 (a)(1) states in part:

> The holder of a license...may (i) make changes in the facility as described in the safety analysis report, (ii) make changes in 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.

10 CFR 50.59 (b)(2) states in part:

> The licensee shall submit...a report containing a brief description of any changes, tests, and experiments, including a summary of the safety evaluation of each.

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

- Increase the probability of occurrence or the consequences of an accident or malfunction of structures, systems, or components important to safety previously identified in the FSAR;
- Create the possibility of an accident of or malfunction of a different type than any preciously evaluated in the FSAR;
- Reduce the margin of safety as defined in the basis for technical specifications;

And therefore, do not involve an unreviewed safety question as defined in 10 CFR 50.59.

JAF-SE-87-160 Rev. 4 REPLACEMENT OF VALVE PACKING WITH STANDARD AND LIVE LOADED FIVE RING GRAPHITE PACKING SYSTEMS

Modification: M1-87-100

The purpose of this modification was to upgrade and improve the reliability of various plant valves that had exhibited recurring packing leakage. This upgrade included the replacement of installed asbestos packing with a "five ring" graphite packing system.

This modification also included the cutting and capping of certain valve leakoff/sealing water lines as these lines are not required with the use of the "five ring" graphite packing system.

The following items were considered in determining the impact of this modification on nuclear safety. The loads applied to the graphite packing systems are typically fifty percent less than those required to obtain the same sealing characteristics as asbestos packing. The weight change associated with the addition of the "five ring" graphite system would not adversely affect the valves' seismic qualification.

In conclusion, no reduction in the margin of safety as defined in the basis for Technical Specifications or change to the Technical Specifications result from the use of graphite packing in place of asbestos packing. Industrial experience has shown that leakage and stem friction loads are reduced and valve stroke times have not increased.

This safety evaluation concluded that this modification does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-88-060 **Rev. 2** **MODIFICATION AND/OR ADDITION OF NON-SAFETY-RELATED PIPE SUPPORTS TO VARIOUS NON-SAFETY-RELATED PIPING SYSTEMS**

Modification: F1-88-067

The purpose of this safety evaluation was to review work associated with modification F1-88-067. This work involved the modification of various damaged and non-functional pipe supports and the addition of missing pipe supports to unsupported, over-spanned, or unrestrained Quality Assurance (QA) Category M (some significance to safety) and QA Category II/III non-safety-related piping systems as defined in the Final Safety Analysis Report (FSAR). It was the intent of this plant modification to provide for the repair of non-functional pipe supports or the installation of missing pipe supports so that the piping systems met the original design requirements.

All pipe supports worked under this modification met the original design code (USAS B31.1.0-1967 Edition through 1969 Addenda - Power Piping Code). There were no changes to the design function of the affected pipe supports or piping systems as a result of this modification. There was no change to the FSAR.

This safety evaluation concluded that the modification and/or addition of pipe supports to QA Category M or QA Category II/III piping systems did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-88-200 Rev. 0 REPLACEMENT OF BARTON MODEL 289 D.P. SWITCHES WITH MODEL 581A-0 D.P. SWITCHES

Modification: M1-86-133

The purpose of this modification was to replace existing ITT Barton Model 289 differential pressure switches with replacement Barton Model 581A-0 switches. The modification provided an environmentally qualified upgrade in the switch and maintained the continued reliability of the subject components.

The scope of this modification included the replacement of eight Model 289 switches with Model 581A-0 switches mounted at the original locations. The process pressure port connections and electrical connections were compatible with the existing instrument tubing and electrical conduit/cable.

The replacement of the switches did not create the possibility of an accident or malfunction of a type different than already described in the Final Safety Analysis Report (FSAR). The replacement switches did not reduce the margin of safety or increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety as previously evaluated in the FSAR.

This safety evaluation concluded that this modification does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-90-046 Rev. 0 UPGRADE LAKE ROAD SERVICE

Modification: F1-89-054

The purpose of this modification was to upgrade the capacity of the East Access Road overhead electrical service line to provide non-essential power to on site support loads.

The scope of this modification included the installation of a 13.2 kilo-volt (KV) outdoor switchgear unit near the intersection of Lake Road and the East Access Road to draw a load of up to 5 milla-volt amperes (MVA) from the Niagara Mohawk Power Corporation Lake Road circuit. The size of the existing East Access Road overhead service line was increased to accommodate up to 12 MVA of future capacity.

This safety evaluation determined that the facilities serviced by this line are used to provide auxiliary services to the plant. The loss of power on the East Access service line would not compromise systems necessary for the safe shutdown of the plant. The implementation of the modification does not increase the probability of occurrence or consequences of an accident or malfunction of structures, systems or components important to safety previously evaluated in the Final Safety Analysis Report (FSAR). No portion of the East Access Road power line nor the auxiliary service facilities served by it are considered in the FSAR.

This safety evaluation concluded that the modification did not involve a change in the Technical Specifications and does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-90-111 Rev. 0 SMALL BORE MANUAL VALVE REPLACEMENT

Modification: M1-90-201

This modification was used to evaluate and authorize the suitable replacement of small bore (2 inch and less) manual valves, check valves, and manual instrument valves with valves of different manufacturers, models, types, and materials. The replacements were required for cases where the original valve was either no longer desired or available.

The modification included a technical evaluation summary sheet to ensure that each valve being considered for use as a replacement received the pertinent technical considerations associated with the valve. Additionally, a Design Input Summary Sheet and an Engineering Change Notice were used for valve replacement approval to ensure the proper level of review and approval for each valve replacement.

Each replacement was individually evaluated in accordance with the requirements of the applicable section of the Final Safety Analysis Report (FSAR) and the Technical Specifications to assure the replacement valve did not change the design function of the valve in the system.

This modification does not affect any system integrity nor involve any changes to the FSAR.

This safety evaluation concluded that this modification does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-90-113 Rev. 0 CIRCULATING WATER SYSTEM FLOW TEST

Modification: M1-90-237

The purpose of this modification was to add test connections on the Circulating Water System piping. Connections were used to perform a test on the Circulating Water System to vary system flow and generate new performance curves for the Circulating Water System pumps. Testing was performed during a plant shutdown period.

The scope of this modification included installing taps and isolation valves at various locations on the system to support temporary dye (Rhodamine WT) injection equipment.

The Circulating Water System is Quality Assurance (QA) Category II/III, non-safety-related. The implementation of this modification does not impact or conflict with any safety-related components or systems or affect overall plant safety.

The dye used for testing was compatible with the piping internals and approved for environmental discharge to the lake by the New York State Department of Environmental Conservation.

This safety evaluation concluded that the modification does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-91-047 Rev. 0 REORIENTATION OF SOUTHWEST CORNER FENCE ALONG WITH INSTALLATION OF FIXED CAMERAS

Modification: F1-91-111

This was a Safeguards Information modification. The safety evaluation may be reviewed upon request.

This safety evaluation concluded that this modification did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-91-136 Rev. 0 VALVE REPLACEMENT PARTS

Modification: M1-91-148

The purpose of this modification was to allow the use of valve spare parts which differ from the descriptions given on plant drawings. These differences involved the materials used to fabricate the spare parts.

Approval for the use of these replacement parts was performed and documented by this modification. Additionally, this modification ensured that the necessary design considerations were evaluated and that the required update of plant documentation systems was completed.

This safety evaluation documents a review to assure that subcomponent changes authorized by this modification did not alter the function of any component or system, did not reduce the margin of safety, or increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report (FSAR). The Technical Specifications were not affected by any subcomponent change.

This safety evaluation concluded that the modification does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-92-003 **Rev. 0** **REORIENTATION OF SOUTHWEST CORNER FENCE ALONG WITH INSTALLATION OF FIXED CAMERAS**

Modification: F1-91-111

This is a Safeguards Information modification. The safety evaluation may be reviewed upon request.

This safety evaluation concluded that this modification did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-92-122 Rev. 1 RWCU PUMP "A" SIDE PROCUREMENT AND INSTALLATION

Modification: F1-90-202

The purpose of this modification was to remove the existing non-functioning "A" side Reactor Water Cleanup (RWCU) pump, 12P-1A, and install in its place a replacement pump similar in design to that installed and operated on the RWCU "B" loop. The replacement was a 50 percent system design flow centrifugal pump for parallel operation thereby restoring the RWCU System to 100 percent of its design flow capacity.

Revision 1 to this safety evaluation was issued to include the following:

1. This modification required a tie into the Control Rod Drive (CRD) System to provide a cooling medium to the 12P-1A seals. After receiving a containment isolation signal pump 12P-1A will trip but the CRD cooling water will continue to flow to the pump seals. To prevent over-pressurization of the seals, operators were instructed to manually shut off the CRD seal water upon a pump trip. The previous revision to the safety evaluation lacked a statement regarding manual operator actions after a pump trip.

2. Engineering Change Notice ECN-F1-90-202-020 made a change to the size/rating of the thermal overload heaters installed on the 12P-1A motor starter. Revision 0 identified the previous thermal overload heater size/rating.

This safety evaluation reviewed the potential impact on plant safety caused by the installation of replacement RWCU pump 12P-1A. The implementation of this modification does not conflict with the design basis of the RWCU System as the original design basis called for parallel two pump operation. The replacement pump and associated pump motor have been purchased/installed to specifications which meet or exceed the original specifications.

Since this modification replaces the defective pump 12P-1A with a pump that has the same design and flow capacity as the original design of 12P-1A, this safety evaluation concluded that this modification does not involve an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-92-182 **Rev. 1** **EVALUATION OF PARTS LOST IN REACTOR CAVITY DURING RELOAD 10 REFUELING OUTAGE**

Modification: **N/A**

The purpose of Revision 0 to this safety evaluation was to examine the potential effects on nuclear safety of parts lost in the reactor cavity which were unaccounted for at the end of Reload 10 refueling outage and therefore were assumed to be in the reactor pressure vessel.

JAF-SE-92-182 Rev. 0 also took credit in its analysis for the Main Steam Line High Radiation monitor initiating closure signals to the Main Steam Isolation Valves (MSIV) in the event of a clad fretting failure.

After the approval of Revision 0 an analysis (Safety Evaluation JAF-SE-94-053, Rev.0) was done which resulted in the removal of the High Main Steam Line radiation closure signal to the Main Steam Isolation Valves (MSIV). This analysis concluded that in the event that fuel cladding failures occurred on a time scale which precluded detection and correction as a result of normal reactor water chemistry monitoring programs, an Offgas System isolation on high radiation will limit the release of radioactive nuclides to acceptable levels.

Continued operation of the Steam Jet Air Ejectors and the Offgas System will be best able to process the moderate release of radioactive nuclides associated with a clad fretting failure.

Revision 1 was generated to remove the credit taken for MSIV closure on a High Radiation Monitor signal and reflect the revised position.

This safety evaluation concluded that this revision did not involve an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-92-184 Rev. 2 CAD SOV REPLACEMENTS

Modification: M1-90-242

The purpose of this modification was to replace the existing Containment Atmosphere Dilution (CAD) System solenoid operating valves which were not rated for use at the design temperature of the CAD enclosure (winter ambient conditions).

The modification replaced the existing model solenoid valves with low temperature ASCO solenoid valves, model X8344G0MF, which have a minimum operating design temperature of -40 degrees Fahrenheit.

This modification does not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety. The modification does not create the possibility of an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report (FSAR). There is no reduction in the margin of safety as defined by the Technical Specifications as a result of this modification.

The replacement of the CAD SOV's per this modification does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-93-001 Rev. 0 EMERGENCY DIESEL GENERATOR VENTILATION SYSTEM ROOM TEMPERATURE ALARMS

Modification: M1-90-206

The purpose of this modification was to enable the Emergency Diesel Generator (EDG) Ventilation System temperature control components to provide temperature indication during normal plant operation when the EDGs are not operating and when the EDG Room supply fans 92FN-1A,B,C,D are secured.

The EDG "Room Temperature Low" annunciators located on local control panels 92HV-9A,B, are set to trip if room temperature decreases below 40 degrees Fahrenheit. The alarms are actuated by EDG Ventilation System control components 92TIS-101A,B,C,D, 92TT-101A,B,C,D, and 92TS-101A,B,C,D. These switches were designed/wired in a fail safe mode (i.e., the alarms will trip both on low temperature and on failure of the alarm switch such as failure of AC power input to the switches). Additionally, these control components were energized through a contact located on the supply fans. During normal plant operation when the EDGs are not operating and the supply ventilation fans were secure, control components were also de-energized.

This modification rewired the 120 volts alternating current (VAC) power feed to the control components 92TIS-101A,B,C,D, 92TT-101A,B,C,D, and 92TS-101A,B,C,D to provide alternating current (AC) power when the supply fans are both operating and shutdown.

The modification did not change the function or operation of the EDG Ventilation System. The margin of safety as defined in the basis of Section 3.11 of the Technical Specifications is not affected.

This safety evaluation concluded that this modification does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-93-006 Rev. 1 SBGT/TRACK BAY FIRE DOOR MODIFICATION

Modification: F1-93-017

The purpose of this modification was to resolve and repair the deficiencies associated with fire door 24R-272-10 located between the Standby Gas Treatment (SBGT) System room and the Reactor Building Track Bay.

The purpose of this door, in addition to providing fire boundary protection, is to serve as a Secondary Containment boundary when the Inner Track Bay Pass door is open. This door is interlocked with both the Pass door and the Inner Track Bay door to prevent an inadvertent breach of Secondary Containment.

The methodology for repairing fire door 24R-272-10 was commensurate with approved and established repair procedures. The procedures addressed the requirements imposed by the Secondary Containment function of the door.

This evaluation determined that the work associated with this modification did not increase the probability of occurrence of an accident or malfunction of structures, systems, or components important to safety previously evaluated in the Final Safety Analysis Report (FSAR). This modification did not create an accident or malfunction of a different type than any evaluated in the FSAR. This modification improved the reliability and operability of SBGT fire door, 24R-272-10.

This safety evaluation concluded that the work associated with Modification F1-93-017 did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-93-019 Rev.0 SOURCE RANGE MONITOR/INTERMEDIATE RANGE MONITOR CABLE REPLACEMENT

Modification: M1-91-015

The purpose of this modification was to replace a section of the prefabricated cables for the Source Range Monitors (SRM) and Intermediate Range Monitors (IRM). This change was initiated by Operating Experience Review Report (OER) P619 (GE SIL 192) which discussed the use of an improved coaxial signal cable that had been found to be successful in reducing electromagnetic interference in the IRMs and SRMs. The improved cable consisted of a triple-shield to replace the old double-shield cable.

The SRM and IRM cable run is divided in two sections; one section inside the drywell and the second section outside the drywell. This modification dealt only with the replacement of cable inside the drywell from the penetration to the detectors.

The replacement of these cables did not affect or change the Final Safety Analysis Report (FSAR) or Technical Specifications. The design change did not affect Plant Security or the Fire Protection System and did not have any adverse environmental impact.

This safety evaluation concluded the modification did not involve an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-93-035 Rev. 0 CONTROL ROOM ACCESS MODIFICATION

Modification: F1-91-085

The purpose of this safety evaluation was to demonstrate the acceptability of the pre-operational test procedure used to verify the operability of the Control Room Door and Security Card Reader installed by Modification F1-91-085. The verification of operability included the proper operation of the new Control Room Access Door installed between the new Administrative Building and the Control Room, the Security Card Reader, and the relocated Emergency Lighting Wall Pack Unit.

The new Control Room door has been designed to maintain the Control Room differential pressure, fire rating, and security rating as specified in procurement specifications.

This evaluation demonstrated that the performance of this pre-operational test would not affect any plant operating system while meeting Security and Fire Protection guidelines.

No design changes were made to affect the differential pressure requirements for the Control Room.

Changes to the Security Plan will be made and presented to the Nuclear Regulatory Commission (NRC) by the Security Department in accordance with NUREG 0908 "Acceptance Criteria for the Evaluation of Nuclear Power Reactor Security Plans".

It was determined from this evaluation that the modification did not involve changes in the plant's Technical Specifications. The probability of occurrence or consequences of an accident or malfunction of structures, systems, or components important to safety previously evaluated in the Final Safety Analysis Report (FSAR) has not been increased because the Control Room boundary does not affect any plant operating system. In addition, the modification did not degrade the security of the plant nor the Fire Detection System.

This safety evaluation concluded that the modification and pre-operational test do not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-93-038 Rev. 0 **CORE SPRAY FULL FLOW TEST LINE PIPE REPLACEMENT**

Modification: M1-93-059

The purpose of this modification was to replace two sections of pipe (lines 14-10"-W23-152-9A,B) located between Core Spray System motor operated valves 14MOV-26A,B and lines 10-16"-W20-152-5A,B. This piping replacement was needed due to cavitation damage from valves 14MOV-26A,B.

The original lines, each approximately ten feet in length of ten inch diameter schedule 40 ASTM A106 Grade B carbon steel pipe and ASTM A234WCB fittings, were replaced with ASTM A 312 type 316L stainless steel pipe and A304 WP316L fittings.

This safety evaluation demonstrated that the above changes are acceptable based on a review of the existing design bases and requirements. The modification did not alter any system function or integrity and lessened the potential of system loss of integrity by increasing the piping resistance to cavitation damage and intergranular stress corrosion cracking. Postulated failure of these lines is enveloped by the analysis in Chapter 14 of the Final Safety Analysis Report (FSAR).

This safety evaluation concluded the modification did not involve an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-93-064 Rev. 1 PRE-OPERATIONAL TEST PROCEDURE POT-12G IMPLEMENTATION

Modification: F1-90-202

The purpose of this safety evaluation was to ensure that the performance of Pre-operational Test POT-12G, "Reactor Water Cleanup Two Pump Parallel Operation" which demonstrates that two pump parallel operation of the Reactor Water Cleanup (RWCU) System at design flow did not present an unreviewed safety question. The test also provided baseline data for the newly installed RWCU pump 12P-1A.

The objective of POT-12G was to provide baseline flow data for RWCU pump 12P-1A and to demonstrate two pump operation of the RWCU System at design flow so that this practice could be incorporated into Plant Operating Procedure OP-28, "Reactor Water Cleanup System".

The design basis for the RWCU System is for two pump parallel operation as described in the Final Safety Analysis Report (FSAR) Section 4.9. The test was prepared using a procedure step derived from the existing OP-28 and the previous operating procedure revisions which provided for two pump operation.

This safety evaluation concluded that the implementation of Pre-operational Test Procedure POT-12G did not affect the margin of safety as defined in the basis for Technical Specifications or the safety analysis of the plant as described in Chapter 14 of the Final Safety Analysis Report (FSAR).

This safety evaluation concluded that this test does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-93-082 Rev.0 PERFORMANCE MONITORING SYSTEM UPGRADE

Modification: F1-86-029

The purpose of this modification was to install new thermocouple assemblies on the main steam and condenser lines located in the Turbine Building. These assemblies are designed to maximize the availability of monitoring the heat balance of the plant.

The scope of this modification also included the installation of terminal boxes, instrument enclosure assemblies and new cables to support the new monitoring capabilities.

This modification is not classified as nuclear safety-related nor does the implementation of this modification impact or interface with any safety-related components or system or affect overall plant safety.

The safety evaluation concludes that the non-safety-related, QA Category II/III classification for the monitoring system was appropriate and the modification does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-93-100 Rev. 1 TEMPORARY DISABLING OF THE AUTOMATIC START CIRCUIT OF CONTROL ROOM SUPPLY FANS 70FN-6A AND 70FN-6B

Modification: M1-93-141

This safety evaluation was the basis for determining whether any safety concerns existed associated with the temporarily disabling the auto-start circuit of Control Room supply fans 70FN-6A and 70FN-6B. A concern arose that the auto-start circuit of the emergency fans may not have met the electrical isolation/physical separation licensing commitments for the plant. In order to prevent a single failure in the automatic start circuit from jeopardizing both emergency fans, the automatic start circuits were temporarily disabled. The auto-start features were restored by Modification M1-94-116 and safety evaluation JAF-SE-94-075, Revision 0.

The Control Room Ventilation System was operated in the normal mode on plant restart following the 1994 Maintenance outage. In the normal mode neither of the emergency fans were in operation. Upon detection of a high radiation level in the Control Room in order to assure that the Control Room would have stayed in a positive pressure environment and had adequate supply of fresh air, the system would have been placed in the isolation mode and the emergency fans would have been manually started in accordance with operating procedures.

As both fans would have been running simultaneously upon receipt of a Control Room ventilation high radiation alarm, the auto-start function of the fans could be temporarily disabled. Running of the two fans would have resulted in a higher flow through the emergency ventilation system components. The components were determined to be adequate to handle the increased flow. Radiological doses to the Control Room operators due to increased flow were evaluated and were found to be below 10 CFR 50, Appendix A, Criterion 19, "Control Room" limits.

This safety evaluation determined that the implementation of this modification was acceptable and did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-93-102 Rev. 0 ADD DRAIN OIL VALVES AND PIPING TO 76P-2(M) ELECTRIC FIRE PUMP MOTOR

Modification: M1-93-146

The purpose of the modification was to provide oil sample lines to the Fire Protection System Electric Fire Pump (76P-2) to facilitate ease of oil sampling.

The scope of this modification included adding 1/2" stainless valves, piping, tubing and end caps to the upper and lower oil drain reservoirs access ports on the Fire Protection System 76P-2 Electric Fire Pump motor to support ease of sampling and tight shutoff.

All equipment installed by this modification is non-safety-related which is consistent with the design requirements of the Electric Fire Pump motor. No safety-related equipment is affected by this modification. This modification does not affect the Technical Specifications or the safety analysis of the plant as described in the Final Safety Analysis Report (FSAR).

This safety evaluation concluded that this modification does not involve an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-93-118 Rev. 2 TEMPORARY OPERATING PROCEDURE TO ALLOW SHUTDOWN OF CONTROL ROOM AND RELAY ROOM VENTILATION SYSTEMS FOR MAINTENANCE

Modification: N/A

This safety evaluation was the basis for determining whether any safety concerns existed associated with a Temporary Operating Procedure (TOP) which allowed the shutdown of the Control Room or Relay Room ventilation systems to allow Electrical Distribution System maintenance activities. These activities were concurrent with modification work on the "A" train of the Relay Room ventilation system which resulted in loss of all Relay Room ventilation. This activity was completed during a plant outage period.

The work activities associated with the temporary shutdown of the Relay Room ventilation were of short duration, noncomplex evolutions. The work activities required the de-energization of the "B" side 10600 vital bus.

The Temporary Operating Procedure provided direction to Operations of actions necessary to shutdown the system, delineated the appropriate Technical Specification requirements, and described the necessary compensatory actions to be taken during system shutdown to ensure that the required system safety functions were maintained.

This evaluation determined that this TOP did not create the possibility for an accident or malfunction of safety-related structures, systems, or components of a different type than previously evaluated in the Final Safety Analysis Report (FSAR). This TOP only allowed system shutdown for a short duration. Procedural controls were specific to ensure that design temperature limits for room air were not exceeded thus ensuring operability of the associated equipment in the area. This TOP did not reduce the margin of safety as defined in the bases for the Technical Specifications because the systems were readily available for manual restart upon an initiating event.

This evaluation concluded that the performance of the TOP to allow temporary Control Room and Relay Room ventilation system shutdown for the performance of identified maintenance activities did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-018 **Rev. 0** **REPLACEMENT OF MAIN STEAM ISOLATION VALVE (MSIV) PILOT SOLENOID VALVES**

Modification: M1-93-079

The purpose of this modification was to replace the existing Main Steam Isolation Valve (MSIV) pilot solenoid valves with solenoid valves manufactured by Automatic Valve Company. The existing valves which were manufactured by ASCO have been discontinued. The replacement solenoid valves are functionally identical to the ASCO valves and have been demonstrated to meet all design requirements for this service.

This safety evaluation documents a review to assure this modification will not affect the capability of the MSIVs to operate as designed. The Final Safety Analysis Report (FSAR) Figure 6.6.2, MSIV Schematic Control Design, was revised to reflect the new solenoid configuration. No change to the Technical Specifications was required.

The safety evaluation concluded that there is no adverse impact on plant safety due to installation of this modification. There is no change in the margin of safety. No unreviewed safety question exists pursuant to 10 CFR 50.59.

JAF-SE-94-027 Rev. 0 CORE SPRAY KEEPFULL MIN FLOW ISOLATION VALVE ADDITION

Modification: M1-94-014

The purpose of this modification was to provide a means to facilitate reverse flow testing of the existing Core Spray (CSP) System keepfull minimum flow check valves (14CSP-62A,B).

The scope of this modification included the installation of two 1" carbon steel isolation gate valves (one isolation valve per check valve). These isolation valves provided plant personnel the ability to verify the operability of the individual safety-related check valves for inservice testing as required by the Second Interval Inservice Testing Program and will allow the exercising and flushing of corrosion from the valve internals.

This piping design and installation were Quality Assurance (QA) Category I, seismic category I. Stress analyses were performed to demonstrate the adequacy of the new pipe configuration. The design function of the Core Spray System was not changed as a result of the modification.

This safety evaluation documents that the Final Safety Analysis Report (FSAR) Core Spray System Figure 7.4-5 was revised to reflect the new pipe configuration.

This safety evaluation concluded that this modification does not involve an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-038 Rev. 2 TEMPORARY POWER REQUIREMENTS DURING BUS OUTAGES

Modification: N/A

The purpose of this safety evaluation was to evaluate the acceptability of installing temporary power feeds to maintain selected station loads energized during outages on 11500, 12500, 11600, and 12600 buses. Revision 2 added an evaluation for installing temporary feeds to maintain power to selected loads on motor control centers (MCCs) MCC-253 and MCC-263 during outages on the 12500 and 12600 buses.

The analysis addressed the adequacy of the equipment used to implement the temporary power feeds and the acceptability of these conditions relative to the requirements defined in Technical Specifications and the Final Safety Analysis Report (FSAR). The loads/buses receiving temporary power were isolated from their normal power source and the loads were considered inoperable. Adequate overcurrent protection and coordination was established for all temporary feed power sources so that no single fault in the temporary loads would have resulted in a loss of the safety-related MCC. The load capacity margin was verified to be adequate and the evaluation determined that miscoordination of the non-safety-related sources would not challenge the safety design basis of the emergency power system.

Providing temporary power as described in this evaluation was determined acceptable because no credit was taken for loads receiving temporary power. Technical Specification operability requirements were met and the temporary loads received adequate electrical protection.

This evaluation concluded that the above described activity did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-049 **Rev. 0** **REPAIR AND UPGRADE OF FIRE DAMPERS**

Modification: F1-91-212

A walkdown of various ventilation systems identified fire dampers and their associated penetrations that required repair/replacement or upgrading in order to comply with 10 CFR 50 Appendix R fire protection requirements or Branch Technical Position BTP9.5-1 Appendix A fire area boundary.

The purpose of this modification was to repair, replace, or upgrade eleven (11) ventilation system fire dampers or fire damper penetrations identified during the inspection.

This safety evaluation reviewed the potential impact on safety-related equipment serviced/protected by these fire dampers during the periods when the dampers were made inoperable for the scheduled repairs or upgrade.

This evaluation determined that the probability of occurrence or consequences of an accident or malfunction of structures, systems, or components important to safety previously addressed in the Final Safety Analysis Report (FSAR) did not increase. The possibility of an accident or malfunction of a different type than any previously evaluated in the FSAR had not been created. This modification did not shutdown, breach or alter any safety-related equipment or heating, ventilation or air conditioning (HVAC) system.

This safety evaluation concluded that the activities and unavailability of ventilation systems and components during fire damper repairs did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-050 **Rev. 0** **12RWC-46 VALVE EQUALIZATION LINE REMOVAL**

Modification: M1-93-007

The purpose of this modification was to permanently remove the 1/2" equalization line from Reactor Water Cleanup (RWC) System manual gate valve 12RWC-46. The socket welded taps associated with the line, located on the valve body and bonnet, were plugged and welded.

The subject equalization line was provided to protect the operation of valve 12RWC-46 against bonnet overpressurization. Bonnet overpressurization occurs when water trapped in the bonnet of a closed valve heats up from the cold condition. 12RWC-46 is a manual block valve which is normally open during system operation. Reactor Water Cleanup System operation is required prior to and during plant startup. Bonnet overpressurization will not occur with the 12RWC-46 valve in the open position during a system temperature increase.

This modification does not involve a change to the Final Safety Analysis Report (FSAR). The equipment removed by the modification will not affect the operation of the Reactor Water Cleanup System nor its response to an accident as described in the FSAR.

This safety evaluation concluded that this modification does not involve an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-053 **Rev. 1** **ELIMINATION OF MSIV CLOSURE FUNCTION AND SCRAM FUNCTION OF MAIN STEAM LINE RADIATION MONITORS**

Modification: F1-93-086

The purpose of this modification was to eliminate the reactor scram function and the Main Steam System isolation valve (MSIV) closure function of the Main Steam Line Radiation Monitors (MSLRM). The need for elimination of these functions was recognized as a generic improvement by the Boiling Water Reactor (BWR) Owners Group who submitted a safety evaluation justifying removal of these functions to the Nuclear Regulatory Commission (NRC) in May 1987 as Licensing Topical Report NEDO-31400. The report was approved by the NRC in May 1991. A Technical Specification amendment request was approved for the James A. FitzPatrick Nuclear Plant in March 1994.

This change reduces the potential for spurious reactor scrams and unnecessary isolation of the reactor vessel. The modification involved only internal wiring changes on Main Control Room panels 09-15 and 09-17.

This safety evaluation reviewed all aspects of this modification from the effect of wiring changes to the various scenarios involved in making this change. The evaluation concluded that this modification did not increase the probability of occurrence or consequences of any accident previously evaluated in the Final Safety Analysis Report (FSAR).

FSAR Chapters 4, 7 and 14 required revision for the implementation of this modification.

This safety evaluation concluded that this modification does not constitute an unreviewed safety question pursuant to 10 CFR 50.59

JAF-SE-94-055 Rev. 1 4160/600V (NON 1E) TRANSFORMER REPLACEMENT

Modification: F1-94-078

The purpose of this modification was to replace the existing General Electric 4160/600 volt (V), 1000 kilo-volt (KVA), Class AA transformers with new 1000/1333KVA, Class AA/FA transformers manufactured by Olsun Electric Corporation.

All transformers replaced by this modification are non-safety-related Quality Assurance (QA) Category II/III transformers feeding non-safety-related QA Category II/III loads.

The higher forced air rating (1333KVA) of the new transformers is not intended for the addition of new load, but to prevent short term overheating and thus extending transformer useful life and reliability.

Transformers 71T-6 and 71T-9 are located in a Category I building and will be seismically supported to prevent damage to adjacent equipment located in the area.

This modification did not increase the probability of occurrence of an accident to the structure, system or components important to safety previously addressed in the Final Safety Analysis Report (FSAR). This modification did not affect any equipment whose margin of safety was defined in the Technical Specifications.

This safety evaluation concluded that this modification does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-064 Rev. 1 10MOV-25A(B) IN-SITU DESIGN BASIS DIFFERENTIAL PRESSURE TEST

Modification: N/A

This safety evaluation was prepared to evaluate Special Test Procedures STP-10BD, Revision 1, "10MOV-25A In Situ Design Basis Differential Pressure Test", and STP-10BE, Revision 0, "10MOV-25B In Situ Design Basis Differential Pressure Test". These STPs were written to stroke Low Pressure Coolant Injection (LPCI) inboard injection valves 10MOV-25A and B from the open to closed position and from the closed to open position during full flow. Valve diagnostic data is collected during valve stroking. The test requires that the Residual Heat Removal (RHR) System be in the Shutdown Cooling mode of operation in the loop which contains the valve being tested.

Revision 1 to this safety evaluation was written to allow performance of the Special Test Procedure with one or both RHR pumps operating.

This evaluation determined that the STPs were performed in full compliance with the applicable Technical Specification for Cold Shutdown reactor conditions. The Emergency Core Cooling Systems (ECCSs) were operable during testing in accordance with Technical Specification Section 3.5.F, ECCS-Cold Condition. In the event that Shutdown Cooling was lost during testing and could not be re-established, Abnormal Operating Procedure AOP-30, "Loss Of Shutdown Cooling" would have been initiated. Loss of Shutdown Cooling is considered an abnormal operating transient and is reasonably expected during the course of operation.

This safety evaluation concluded that the performance of the Special Test Procedures during reactor cold shutdown conditions did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-065 Rev. 0 REMOVAL OF MAIN TURBINE TURNING GEAR MOTOR FAST SPEED

Modification: M1-94-083

The purpose of this modification was to resolve the problem of the Main Turbine Generator rolling off the turning gear when the turning gear motor changes from low speed to fast speed.

The scope of this modification was to eliminate the high speed function of the turning gear motor. Specifically, the thermal overload elements were removed from the fast speed starter in motor control center 71MCC-262-0C3 permanently disconnecting the fast speed motor winding from the 600 volts alternating current (VAC) power supply.

The fast speed elimination does not affect the function, performance or reliability of the turbine turning gear to provide uniform cooling, controlled starting and constant oil film lubrication. Its elimination will provide stability to the turbine start-up operation by preventing the turbine from rolling off the turning gear when the turning gear motor changes speed from slow to fast.

The equipment associated with this modification is located in a mild environment (Electrical Bay 272' elevation and panel 09-7) and therefore did not affect the Environmental Qualification Program. It also did not affect the Fire Protection, Quality Assurance (QA), or Security programs. There is no change to the Final Safety Analysis Report (FSAR) and no reduction in the margin of safety to the plant.

This safety evaluation concluded that this modification does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-071 Rev. 1 UFSAR CHANGE REQUEST FOR THE STORAGE OF QA RECORDS

Modification: N/A

This safety evaluation was the basis for determining whether any safety concerns existed associated with a change to the Final Safety Analysis Report (FSAR) that provided for the interim storage of quality assurance records. It had been established in this safety evaluation (Revision 0) that interim storage of quality assurance records in one hour fire rated cabinets meets the requirement for record storage required by the National Fire Protection Association (NFPA) fire codes. A Deviation Event Report was written because Nuclear Regulatory Commission (NRC) Regulatory Guide 1.88 was removed as a reference from the FSAR since it was considered obsolete by the NRC. Regulatory Guide 1.88 endorsed American National Standards Institute (ANSI) Standard N45.2.9-1974 for the storage of quality assurance records. To satisfy concerns published in the Deviation Event Report, Revision 1 to this safety evaluation reinstated the reference to NRC Regulatory Guide 1.88.

This evaluation determined that the change to the FSAR involving the reinstatement of reference to Regulatory Guide 1.88 did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-075 Rev. 0 RESTORATION OF AUTO-START FUNCTION FOR 70FN-6A AND 6B

Modification: M1-94-116

The purpose of Modification M1-94-116 was to restore the auto-start (original design) function for the Control Room Emergency Ventilation Fans (70FN-6A and 70FN-6B) and to correct deficiencies associated with cable separation in the fan control circuits.

The installation of interposing relays for each fan control circuit (auto-start initiation) eliminated the single failure concern associated with fan starter relays. The re-identifying and re-routing of the cables and re-identifying of raceway pertinent to the fan auto-start function ensured that proper separation between trains "A" and "B" was maintained throughout the control circuits.

This evaluation determined that restoring the Control Room Emergency Ventilation Fan's auto-start initiation function and correcting identified cable separation concerns did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety. These actions did not create the possibility of an accident or malfunction of a different type then previously evaluated in Section 14 of the Final Safety Analysis Report (FSAR) and did not reduce the margin of safety as defined in Section 3.11 of the Technical Specifications.

This safety evaluation concluded that the performance of Modification M1-94-116 did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-078 Rev. 0 FIRE PROTECTION - EMERGENCY SERVICE WATER CROSS-TIE

Modification: M1-88-238

The purpose of this modification was to provide an alternate method for supplying cooling water to the Emergency Diesel Generators (EDG) utilizing the Emergency Service Water (ESW) System piping and the Fire Protection (FP) System fire pump as the motive force This was accomplished with a removable cross-connection between the ESW System and the FP System. The removable cross connection consists of a 4" fire hose dedicated for this application.

The use of this system would allow a FP diesel fire pump to provide approximately 1100 gallons per minute (gpm) of low pressure cooling water to the EDGs by way of the ESW piping in the event the EDGs are required to be operable and the ESW System pumps are inoperable.

The respective FP-ESW cross-tie valves, fittings and associated piping meet the Quality Assurance (QA) Category and seismic class requirements for both the FP and ESW Systems. This modification does not adversely affect the operation or function of either the FP System as described in the Final Safety Analysis Report (FSAR) Section 9.9 or the ESW System as described in the FSAR Section 9.7. The modification does not affect the Technical Specifications.

This modification was committed to the Nuclear Regulatory Commission in the James A. FitzPatrick Nuclear Power Plant Individual Plant Examination (IPE).

This safety evaluation concluded that this modification does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-079 Rev.4 SRV LEAK DETECTION THERMOCOUPLE HIGH TEMPERATURE ALARM SETPOINT CHANGE

Modification: N/A

The purpose of this evaluation was to determine the acceptability of the Safety Relief Valves (SRVs) tailpipe temperature alarms and the proposed change in the alarm setpoint for SRVs "D", "E", and "H" with respect to the values given in Table 7.4-2 of the Final Safety Analysis Report (FSAR). The alarm setpoint given in the FSAR was 200 degrees Fahrenheit. Revisions 0 and 1 of this safety evaluation accepted operation with the current setpoints for these alarms at 300 degrees Fahrenheit for all SRVs with the exception of SRVs "E" and "L" based upon temporary modifications TM-94-112 and TM-94-193, "Raise Setpoints of "E" and "L" SRV Tailpiece Temperatures to 315 degrees Fahrenheit Due to Leaking SRVs".

This safety evaluation demonstrated that there were valid technical reasons for the discrepancy between the FSAR setpoint and the required setpoints. Data taken during the last two operating cycles indicated that the baseline temperatures for the thermocouples in their current locations at full power ranged from 140 to ≈300 degrees Fahrenheit. In their current locations a 200 degree Fahrenheit alarm setpoint would be unreasonable for those SRVs with baseline temperatures near or above 200 degrees Fahrenheit.

This evaluation also demonstrated that the continued operation with the high temperature alarm setpoints for SRVs "D", "E", and "H" set to 324 degrees Fahrenheit and the setpoint for the remaining SRVs set at 300 degrees Fahrenheit was acceptable.

The purpose of the SRV tailpipe thermocouple is to provide leak detection and backup valve position indication for the SRVs. This evaluation has shown that the thermocouples were performing these functions.

An FSAR change was submitted to reflect the current tailpipe thermocouple temperature alarm setpoint changes.

Revision 4 to this safety evaluation was generated to review and provide assurance that with the increased alarm setpoint values the functions of the thermocouples (provide leak detection and backup valve position indication) were maintained.

This safety evaluation concluded that the increased alarm setpoint values did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-081 Rev. 1 EDG AIR START SOLENOID VALVE FILTER ADDITION

Modification: M1-87-069

The purpose of this modification was to add solenoid valve filter assemblies to the Emergency Diesel Generator (EDG) Air Start System solenoid valve supply lines. The equipment is designed to provide filtered air to the EDG Air Start solenoid valves and reduce the chances of foreign materials entering the valves and causing the valves to stick open.

The scope of this modification included the installation of filters with bypass check valves to the air supply lines. The purpose of the check valve bypass line is to allow supply air to the Air Start solenoids in the event of filter clogging. The equipment is installed in 3/8 inch Type 304 stainless steel tube lines which run from the Air Start System inlet piping to the EDG Air Start solenoid valves. The new tubing meets the material and design rating for the EDG Air Start System.

This modification did not change the function or adversely affect the ability of the EDG Air Start System to start and carry a load within 10 seconds as described in Section 8.6 of the Final Safety Analysis Report (FSAR). This modification did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR.

This safety evaluation concluded that this modification does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-082 Rev. 0 **DISABLING OF 29MOV-102 (OP)**

Modification: M1-91-102

The purpose of this modification was to electrically disable Main Steam (MS) System motor operated valve 29MOV-102. The subject valve is located on the reactor head vent line between the vessel head and the inboard "A" Main Steam isolation valve (MSIV) 29AOV-80A.

The scope of this modification included disconnecting and classifying as spares all electrical equipment involved in the subject MOV electrical circuit.

Isolation of the reactor head vent line is required during specific test evolutions such as Reactor Pressure Vessel (RPV) hydro, leak tests, and times where it is necessary to flood the RPV. To preclude degrading the valve of its isolation function, 29MOV-102 was electrically disabled in the open position but manually operable.

This modification did not introduce any change to the reactor head vent valve (29MOV-102) function. The disabling of the valve's motor operator did not increase the probability of occurrence or consequences of any accident or malfunction of equipment important to safety and did not create the possibility of an accident or malfunction of a different type than previously analyzed.

The safety evaluation documents that the Final Safety Analysis Report (FSAR) Main Steam System Flow Diagram, Figure 4-11.1, was revised.

This safety evaluation concluded that this modification does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-089 Rev. 1 ORGANIZATION CHANGES IN NEW YORK POWER AUTHORITY MANAGEMENT STRUCTURE AND THE NUCLEAR GENERATION DEPARTMENT

Modification: N/A

The purpose of this safety evaluation was to describe and evaluate proposed changes to the overall management structure of the New York Power Authority and the management structure of the Nuclear Generation Department as described in the Final Safety Analysis Report (FSAR).

The purpose of Revision 1 to the safety evaluation was to:

- Modify the title 'Executive Vice President and Chief Nuclear Officer' to read 'Chief Nuclear Officer' as a result of action taken at the December 15, 1995 Trustees Meeting.

- Define the new business units as approved at the same meeting. The approved organization structure consists of six business units instead of five as described in the initial issue of this safety evaluation.

- Update the organization of the Nuclear Generation Department as stated in New York Power Authority memoranda WJC-95-011, 'Organization of Nuclear Generation Department' dated 1/23/95.

A Technical Specification amendment was submitted and FSAR Sections 13.2 and 17.2 were also revised to reflect the organization changes.

This evaluation concluded that the changes were administrative in nature with no effect on nuclear safety and that the changes did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-092 Rev. 0 INSTRUMENT AIR SYSTEM IMPROVEMENTS

Modification: M1-91-093

The purpose of this modification was to enhance and improve the reliability of the Instrument Air System (IAS). This modification provided for the installation of a check valve, additional isolation valves, and a bypass line downstream of the air dryer after-filters. These improvements will prevent the possibility of Instrument Air System depressurization due to the loss of power to the air dryers and subsequent air flow through the purge valves if the system air compressors failed. Additionally, the modification increased the design temperature of the piping downstream of instrument air dryer after-filters to 450 degrees Fahrenheit at 150 pounds per square inch gauge (psig) in order to mitigate air dryer malfunctions.

The implementation of this modification did not increase the probability of occurrence or consequences of any accident or malfunction of equipment important to safety as described in Chapter 14 of the Final Safety Analysis Report (FSAR). The new piping and components will not affect the function or operation of the existing Breathing, Instrument, and Service Air Systems in any way as described in the FSAR.

This safety evaluation concluded that this modification does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-094 **Rev. 0** **1" DIAMETER CRD UNIT FLANGE CAP SCREW CHANGE**

Modification: **M1-92-319**

The purpose of this modification was to evaluate the acceptability of changing the material and the design of the cap screws that secure the Control Rod Drive (CRD) mechanism main flange to the drive housing and to evaluate the use of a redesign of the CRD flange cap screw washer. The design changes by the Nuclear Steam Supply System (NSSS) vendor (General Electric) were to address CRD cap screw crack indications found at other General Electric designed boiling water reactors during CRD maintenance (reference General Electric SIL 483).

The new cap screw material is AISI 4340 which has a higher yield and tensile strength when compared to the original AISI 4140 material. Additionally, the new cap screw design has a larger radius at the head to shank transition region which will act to reduce stress concentrations in this region. The new washer is slotted to facilitate draining.

The loss of coolant which would result from a CRD flange bolt failure is enveloped by the existing Loss of Coolant Accident (LOCA) analysis in Section 14.6 of the Final Safety Analysis Report (FSAR). The CRD cap screws (bolts as mentioned in the FSAR) are discussed in Section 3.5 of the FSAR. This section of the FSAR was revised to reflect the higher strength that is being allowed. The use of the new cap screws does not impact the existing analysis or increase the consequence or probability of occurrence of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. Applicable design codes are satisfied (ASME Section III) and are invoked (ASME Section XI) for installation and inspection.

This safety evaluation concluded that this modification does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-102 **Rev. 0** **RAISING THE HYDROGEN ADDITION FLOW RATE UP TO 19 SCFM**

Modification: **N/A**

This safety evaluation determined if raising the hydrogen addition rate to 19 standard cubic feet per minute (SCFM) constituted an unreviewed safety question. General Electric had recommended that hydrogen injection be increased from approximately 13.6 SCFM to approximately 18 SCFM to fully protect the Recirculation System stainless steel piping from Intergranular Stress Corrosion Cracking (IGSCC). Based on the existing available hydrogen injection system, the expected maximum achievable flow rate with three train injection in automatic was between 18 and 19 SCFM.

The following effects of raising the hydrogen injection flow rate were addressed:

- As hydrogen injection increases, main steam radiation levels increase. The main Steam Line High Radiation Monitor alarm and trip setpoints had to be changed to account for the higher injection rate.
- The increase in the hydrogen injection rate would have no adverse safety concerns involving the Hydrogen Leak Detection System.
- The increase in the hydrogen injection rate would not create or pose any safety concerns for equipment associated with the Main Steam Lines and Main Condenser.
- Offgas System recombiner temperatures would be lowered, however, a Final Safety Analysis Report (FSAR) change was made to reflect revised oxygen flow rates.
- Potential higher hydrogen rates to the Torus through the Safety Relief Valves did not pose a safety concern to the Torus.
- Higher hydrogen levels in Containment following a Loss Of Coolant Accident (LOCA) would not create a safety concern.
- Onsite dose rates were evaluated with compensatory actions implemented in areas to control increased radiological doses. Offsite dose rates were not effected.

The radiological effects and changes to the Main Steam Line Radiation Monitor setpoints had no effect on the safety design basis of the plant.

This evaluation determined that the increase in the hydrogen injection flow rate did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-115 Rev. 1 REVISED QA CATEGORY OF VARIOUS REFUELING EQUIPMENT IN SYSTEM 008

Modification: N/A

The purpose of this safety evaluation was to provide justification for changing the Quality Assurance (QA) category of various refueling equipment components.

Specific Refueling Bridge equipment and tools that were listed as QA category I (safety-related) were changed to QA category M (some significance to safety). QA category M components are those that:

- Prevent inadvertent criticality during Refueling operations,
- Prevent withdrawal of more than one control rod except during spiral unload or shutdown margin testing,
- Are associated with Refueling interlocks,
- Include Refueling equipment directly associated with handling spent fuel whose failure could initiate a refueling accident,
- Include Refueling Bridge structural steel and weight bearing components.

None of the included components are required to prevent or mitigate the consequences of a refueling accident due to the fact that the Final Safety Analysis Report (FSAR) design basis accident (DBA) assumes that the accident has already occurred. Based upon this assumption and that the consequences of the accident are not mitigated by the refueling equipment, the components were classified as non-safety-related.

This evaluation reviewed the potential impact on plant safety due to the Refuel Bridge component classification changes and determined that this did not conflict with the design bases for the definition of safety-related components, systems or structures as stated in the FSAR. The change did not have any adverse impact on any other safety-related or environmentally qualified structures, systems, or components and had no impact on Technical Specifications.

This safety evaluation concluded that changing the QA category of specific Refueling Bridge equipment and tools did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-117 Rev. 2 ADDITION OF FIRE HOSE STATIONS REQUIRED BY AMENDMENT 47 INTO AP-01.04 AND DELETION OF DRIP CONNECTIONS REQUIREMENTS

Modification: N/A

This safety evaluation was the basis for determining whether any safety concerns existed associated with changes to Administrative Procedure AP-01.04, "Tech. Spec. Related Requirements, Lists and Tables". The list and table changes involved the addition of the fire hose station installed under Modification F1-79-008 and the deletion of fire hose stations that once provided coverage in areas where the hose stations installed by the modification now provide coverage.

Revision 2 to this safety evaluation evaluated the change to the Final Safety Analysis Report (FSAR) Section 9.8.3.1.3 concerning the use of drip connections on interior fire hose stations, specifically, the removal of this paragraph from the FSAR in its entirety. Drip valves remain on many interior fire hose stations, however, removal of the statement in the FSAR allowed physical removal of drip valves on the hose stations on an as-needed basis.

This evaluation determined that this revision to the FSAR did not change the function, performance, or integrity of boundaries that form or support the primary protective barrier (fuel cladding, primary pressure or containment) on which the consequences of a previously identified accident are based. This activity did not contribute to any failure modes analyzed in the FSAR nor did it affect a system such that a new failure mode was created. This revision did not degrade or prevent any action assumed to occur in the FSAR that mitigate the consequences of a malfunction of safety-related structures, systems or components.

This safety evaluation concluded that this revision to the FSAR did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-120 Rev. 1 STATION AND LPCI BATTERY MODIFIED PERFORMANCE TESTS

Modification: N/A

This safety evaluation was the basis for determining whether any safety concerns existed associated with the proposed alternate Station and Low Pressure Coolant Injection (LPCI) System battery test method described in Maintenance Procedures MST-071.24, "Station Battery B Modified Performance Test", MST-071.25,"LPCI Battery Modified Performance Test", and MST-071.26,"Station Battery A Modified Performance Test". These procedures were intended to be used when both the battery service and performance tests were required to be performed consecutively. This testing method allowed a single modified performance test to be utilized which met the Technical Specification requirements for both tests by ensuring that the batteries were capable of supplying the design basis duty cycle loads and by performing a discharge test (discharge) to determine battery capacity and provide trendable data. The implementation of this single test increased the availability of the respective batteries because of the reduced time required for testing and charging, provided the acceptance criteria was met.

Battery equipment configuration and required plant conditions during the test were equivalent. Technical Specification surveillance requirements were met because the modified performance tests encompass the service and performance test requirements of the Station and LPCI Batteries and are more conservative than the individual service and performance tests.

The safety evaluation concluded that the performance of the Station and LPCI Battery modified performance tests did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-122 **Rev. 0** **MAIN TURBINE VIBRATION TRIP LOGIC MODIFICATION**

Modification: M1-94-171

The purpose of this modification was to install a programmable controller in the Main Turbine High Vibration Trip circuit with the capability of reducing the probability of Turbine and subsequent reactor trips caused by single component equipment protection system actuation. The reduction of the probability of spurious trips serves to mitigate unnecessary challenges to Reactor Protection Systems (RPS) and the plant in general.

The modification protects against single component failure, provides time for plant operators to evaluate a single bearing high vibration signal and, as necessary, bring the unit off line in a controlled manner. It also ensures automatic Turbine trip if a vibration condition is severe enough to propagate to another bearing.

The interfacing system associated with the modification is not safety-related. No changes to the Technical Specifications or the Final Safety Analysis Report (FSAR) are required as a result of this modification. The implementation of this modification does not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. The Main Turbine trip on high vibration is not a precursor to, and does not affect the design input/assumptions for the modeling of accidents addressed in the FSAR.

This safety evaluation concluded that this modification does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-123 Rev. 0 EVALUATION FOR THE TEMPORARY USE OF BWNT AUXILIARY WORK BRIDGE DURING THE CORE SHROUD REPAIR PROJECT

Modification: N/A

This evaluation provided a review of the safety issues associated with temporarily locating an "Auxiliary Work Bridge" on the Refuel Floor of the Reactor Building. The work bridge was used during periods of normal in-vessel inspection and core shroud repair activities. The "Bridge" consisted of major sub-assembled components of welded construction which were bolted together and made temporarily available on the existing refuel bridge rails.

The scope of the evaluation was limited to the following three areas:

- The structural qualification of the "Auxiliary Work Bridge" over the Reactor Cavity; also, the safe load path and rigging procedure for movement of the bridge after assembly,
- A seismically acceptable staging location on the northwest corner of the building,
- Structural qualification of the refueling bridge rails.

Administrative procedures and precautions were developed and used in work instructions for activities associated with the "Auxiliary Work Bridge".

The evaluation determined that the design, rigging paths, load paths, lost parts consideration, and seismic considerations were met and that the temporary use of the "Auxiliary Work Bridge"was an acceptable procedure.

The probability of occurrence or consequences of an accident evaluated in the Final Safety Analysis Report (FSAR) were not increased since the handling of heavy loads of this magnitude is performed periodically with all the administrative controls of Maintenance Procedure MP-88.1, "Heavy Load Handling" which is in compliance with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants". The probability of an accident or malfunction of a different type other than previously evaluated had not been created because this operation involved a routine task of moving heavy loads on the Refuel Floor. The margin of safety as defined in the bases of Technical Specifications were not reduced because adequate precautions were exercised to not cause overload conditions. The stress in all members were within applicable code allowances.

This safety evaluation concluded that the temporary use of the "Auxiliary Work Bridge" during core shroud repairs did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-124 Rev. 1 REACTOR CORE SHROUD REPAIR

Modification: F1-94-036

In response to General Electric (GE) SIL 572, Revision 1, "Core Shroud Cracks", and United States Nuclear Regulatory Agency Generic Letter 94-03, "Intergranular Stress Corrosion Cracking (IGSCC) of Core Shrouds in Boiling Water Reactors", a review of the fabrication and operational history of James A. FitzPatrick Nuclear Power Plant was conducted to determine the potential for IGSCC in the core shroud circumferential welds. The review concluded that weld numbers H1 through H7 at the James A. FitzPatrick Plant may be susceptible to IGSCC.

The scope of this modification included the repair assembly description, repair installation activities including mockups, tool qualification, and training. The repair consisted of ten (10) stainless steel tie-rod/radial restraint assemblies which were installed in the shroud/reactor vessel annulus between attachment points near the top of the shroud and the lower shroud support plate/gussets structure.

This safety evaluation addressed all activities/operations during the installation of this modification. Additionally, this safety evaluation documents the changes to the Final Safety Analysis Report (FSAR) based on the actual completed modification work.

The evaluation concluded that the basic physical configuration and function of the core shroud were not changed by the modification and the repaired shroud continues to meet the structural requirements of the applicable design codes. The repair ensures that the shroud, even if cracked, will perform its safety function. The repair had no impact on overall Emergency Core Cooling System performance. The repair assembly and the repaired shroud satisfy all of the applicable FSAR criteria including earthquake, vibration, and environmental effects (e.g. radiation, temperature and corrosion).

This safety evaluation concluded that the core shroud repair activities did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-127 Rev. 0 CYCLE 12 CORE RELOAD

Modification: M1-94-164

The reactor core was reloaded into a new loading pattern for operation in Cycle 12. The fresh reload fuel, a total of 204 assemblies, are of two types; 200 each General Electric Company GE11 and 4 each Siemens Power Corporation ATRIUM-10A lead fuel assemblies. The Core Operating Limits Report (COLR) is revised as necessary to include the new fuel bundle limits and cycle specific operating limits.

Refuel of the reactor has replaced 204 assemblies and shuffled the core to a new configuration. General Electric is the design organization responsible for the engineering support of the core configuration. Fuel assemblies were replaced to allow for a sustained period of power generation consistent with James A. FitzPatrick Nuclear Power Plant goals.

This safety evaluation documents the changes made to the reactor core by refueling prior to the start of operating Cycle 12. The safety analyses performed on the Cycle 12 core were reviewed and found adequate in scope and content against the requirements in General Electric Licensing Topical Report, General Electric Standard Application for Reactor Fuel (GESTAR), Technical Specifications, and the Final Safety Analysis Report (FSAR). The safety analyses properly supports the limits established in the COLR.

This safety evaluation concluded that the operation of the FitzPatrick Plant in the Cycle 12 core to the limits given in the COLR does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-132 Rev. 1 RESTORATION OF THE RELAY ROOM SYSTEM TO NORMAL MODE

Modification: N/A

The purpose of this safety evaluation was to determine if taking the Relay Room Ventilation System (RRVS) out of the isolation mode and returning it to the normal operating mode constituted an unreviewed safety question.

The Relay Room Ventilation System had been placed in the isolation mode as a result of concerns regarding the potential for not meeting electrical cable separation requirements. Specifically, RRVS isolation valves MOV-105 (intake valve) and MOV-106 (exhaust valve) are redundant to RRVS dampers MOD-100 (intake damper) and MOD-103 (exhaust damper). MOV-105 has three cables routed in the 'red' system, MOV-106 has two cables routed in the 'blue' system and one cable routed in the 'red' system. This setup presents a channelization problem since this configuration does not comply with the Final Safety Analysis Report (FSAR).

No immediate operability concern existed since, in the isolation mode, the isolation valves are closed and the RRVS is capable of performing its safety function.

This evaluation reviewed licensing and design basis information relative to the RRVS. The evaluation determined that manual operation of the RRVS isolation valves was an acceptable alternative to remote manual operation of these valves and is consistent with the plant licensing and design basis. This evaluation further concluded that since these valves may be operated manually the identified cable routing problems were not outside the licensing and design basis.

This safety evaluation also examined and clearly defined the licensing and design basis of the RRVS isolation valves with special emphasis on the remote-manual function (i.e., electrical operation from the Control Room) of these isolation valves.

This safety evaluation concluded that taking the RRVS out of the isolation mode and returning it to the normal mode did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-133 Rev. 1 SUPPLYING REACTOR WATER CLEANUP PUMP SEALS FROM CONDENSATE TRANSFER

Modification: N/A

Reactor Water Cleanup (RWCU) System pump seals are normally supplied seal purge water from the Control Rod Drive (CRD) Hydraulic System. During plant outages the CRD Hydraulic System may be secured for maintenance when the RWCU System pumps are required to be in operation to maintain reactor water chemistry and water purity during in-vessel work. During these periods a temporary means of providing seal water is required.

The evaluation justified the acceptability of a temporary modification which allowed the Condensate Transfer System to be used as a source for RWCU System pump seal purge water during periods when the CRD Hydraulic System was unavailable.

The RWCU Systems has various safety-related functions.

This evaluation determined that the implementation of this temporary modification did not increase the probability of occurrence or consequences of a malfunction of equipment important to safety evaluated previously in the Final Safety Analysis Report (FSAR) because the containment isolation valves of the RWCU System were not affected. The only safety-related equipment relative to both the Reactor Pressure Vessel (RPV) coolant inventory and this modification were the containment isolation valves. These valves are not affected by the normal seal purge water supply and/or this temporary modification.

This safety evaluation concluded that this temporary modification did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-135 **Rev. 0** **OPERATION OF THE CO2 PELLET RADIOACTIVE DECONTAMINATION DEVICE**

Modification: **N/A**

This safety evaluation was prepared to evaluate the use of a device to decontaminate radioactive material by the use of sprayed carbon dioxide pellets. Use of the cleaning unit provides improved capability for decontaminating tools, hardware, and other components.

The CO2 pellet cleaning unit consists of a 14 ton liquid storage tank, a compressed air delivery system, a CO2 pelletizer, a delivery gun, a cleaning enclosure, and a support equipment enclosure. The unit is self-contained and is portable.

The system cleans radioactively contaminated tools and other hardware by means of high velocity delivery of CO2 pellets. Liquified CO2 is expanded in a controlled manner to form CO2 snow, that is in turn compressed into small cylindrical pellets. A gun directs the pellets at the object to be cleaned.

The pellet cleaning station is not designed or used to collect, treat, or dispose of radioactive waste. The system is not a Radioactive Waste Management System or a Radwaste System. The device is not important to nuclear safety. The operation of the equipment does not affect safety related equipment because there is no interface with plant equipment. A review of the system design showed that the device was constructed using generally accepted design principles for systems containing radioactive waste. The most likely failure would be a break in both the equipments HEPA filters and a subsequent release of the entrained radioactive material. In the event of a HEPA filter failure and a significant release of radioactivity an air monitor would rapidly alert the operators of the unit that a release had occurred.

A review of the plant's New York State Pollution Discharge Elimination System (SPDES) revealed no effects to the plant's specified environmental releases.

This safety evaluation concluded that the use of the CO2 pellet radioactive decontamination device did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-139 **Rev. 0** **EVALUATION OF M.G. SET ROOM PLUGS REMOVAL FOR ACCESS TO STEAM TUNNEL DURING PLANT OPERATION**

Modification: **N/A**

The purpose of this safety evaluation was to evaluate the significance of removing one or more of the Steam Tunnel removable concrete floor plugs in the Motor Generator (M.G.) Set Room to provide a secondary access to the Main Steam Tunnel during plant operation.

The additional access location provides plant personnel with a quick access to the outboard Main Steam Isolation Valves (MSIVs) and to the Feedwater System outboard check valves. This additional access path reduced the exposure of personnel to gamma radiation associated with the Main Steam Lines and reduced the time and distance to travel in the Main Steam Tunnel. This access also significantly reduced personnel exposure to high ambient temperatures and provided easier emergency egress when required.

This evaluation determined that the removal of one or more of the Steam Tunnel removable concrete floor plugs in the M.G. Set Room during plant operation does not increase the probability of occurrence of an accident evaluated in the Final Safety Analysis Report (FSAR) because the FSAR identifies and evaluates all accident conditions including the Main Steam Line break outside of secondary containment and controls are specified to ensure there are no unmonitored radiological releases as a result of this activity. The change in the release pathway of radioactivity to the atmosphere (from the M.G. Set Room and Turbine Building instead of the Turbine Building alone) will not alter the potential consequences of the design basis accident. This plug removal did not create the possibility of a malfunction of equipment important to safety of a different type than evaluated previously in the FSAR because there is no safety-related equipment in the M.G. Set Room.

This safety evaluation concluded that the completion of the activity described above did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-140 Rev. 0 ALLOWABLE FUEL BUNDLE CONFIGURATIONS AROUND FUEL PREPARATION MACHINES

Modification: N/A

The purpose of this evaluation was to evaluate permissible fuel configurations in and around the fuel preparation machines in the Spent Fuel Pool outside the spent fuel racks.

Reactor analyst fuel handling procedures allow up to two fuel bundles to be in or around a fuel preparation machine (FPM) if this configuration is separated from the spent fuel racks by 12 inches or more. Prior to the installation of the Holtec racks this required separation was ensured because the existing spent fuel racks were greater than 48 inches from the FPM. When the Holtec spent fuel racks were installed to the east of the existing racks it became possible for a second bundle to be in or around the FPM without meeting the separation requirement.

This evaluation determined that fuel bundle configurations permitted in the fuel handling procedures were acceptable with the Spent Fuel Pool geometry changes made by the installation of the Holtec fuel racks. Specifically, the decreased distance between the fuel preparation machine and the Holtec fuel racks does not significantly increase the reactivity in the Spent Fuel Pool during normal and abnormal conditions as required by the Technical Specifications. The accident previously evaluated in the Final Safety Analysis Report (FSAR) related to this change is the Refueling Accident in which a bundle is dropped from its maximum height onto other fuel in the core. A fuel assembly drop in the pool is enveloped by the drop in the core because the bundle is dropped from a lower height.

This safety evaluation concluded that the fuel bundle configurations described above did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-94-142 Rev. 1 RWR SYSTEM ECP FLANGE REPLACEMENT

Modification: M1-95-001

During the 1992 Refuel Outage Electrochemical Potential (ECP) probes were installed in the Reactor Recirculation (RWR) System pump suction piping to verify ECP measurements made using the Crack Arrest Verification System (CAVS). The objectives for originally installing the ECP probe and its associated modified blind flange, which was used to house the ECP measuring instrument, have been met. Results have shown that the plant needed to increase the amount of hydrogen flow into the RWR System to fully protect system piping.

The purpose of this modification was to replace the modified blind flange used to house the ECP probes and to disconnect probe cables and to spare them in place. The replacement blind flange (ASTM A182 F304) is similar to the existing flange (ASTM A182 F304L).

The flange installed by this modification meets the material and design rating for the RWR System. The disconnecting and sparing of the probe cables will not affect any other plant safety system or equipment as defined in the Final Safety Analysis Report (FSAR). This modification does not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR.

This safety evaluation concluded that this modification does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-001 Rev. 0 REMOVAL OF SIGHT GLASSES FROM RWCU HEAT EXCHANGER VENT AND DRAIN PIPING

Modification: M1-95-004

The purpose of this modification was to remove three sight glasses from the vent and drain piping in the Reactor Water Cleanup (RWC) System regenerative heat exchangers 12E-2A,B. The sight glasses were not used during plant operation and required corrective maintenance to repair water leaks. The sight glasses were replaced with sections of piping constructed to the same specifications as other RWC System piping. In the event that operators should have the need to determine if flow is occurring in the vent and drain line temperature probes can be used.

This piping design and installation are Quality Assurance (QA) Category II/III. The RWC System regenerative heat exchangers do not perform any function important to safety and the replacement of their vent and drain sight glasses does not affect any equipment evaluated in the Final Safety Analysis Report (FSAR).

A change to the FSAR Figure 4.9-1 will be performed to remove the sight glasses from the RWC System - Piping and Instrument Diagram.

This safety evaluation concluded that this modification does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-002 Rev. 0 INSTALL TEMPORARY FLOW METER AND DISCONNECT THE AUTOMATIC FUNCTION OF THE FLOW CONTROL VALVE OF THE REACTOR BUILDING EXHAUST RADIATION MONITORING SAMPLE SYSTEM

Modification: N/A

The purpose of this safety evaluation was to evaluate the acceptability of installing a temporary flow indicator and test connection in the Reactor Building Exhaust Monitoring Sampling System, to disconnect the sample automatic flow control function (flow control valve), and to disconnect the attendant Control Room indication of low flow through the Emergency and Plant Information Computer (EPIC). The permanently installed flow indicators that normally indicate and control flow had experienced operation and reliability problems.

The safety-related functions of the Reactor Building Ventilation Exhaust Radiation Monitor System are to initiate Standby Gas Treatment (SGT) System, isolate the Reactor Building Ventilation System, and isolate Containment Vent and Purge System. The initiating signal for these systems is a result of the gaseous monitor.

The primary function of the existing flow indicator is to provide local indication and regulate the flow through the monitoring system by changing the position of the system's bypass valve.

This function was temporarily re-established by installing a rotameter and associated materials. The proper flow was achieved by manually adjusting and monitoring flow on operator rounds. Its secondary function is to provide a signal to the Emergency and Plant Information Computer (EPIC) that system flow is low. This function was accomplished by operators performing rounds and observing flow once per shift.

This evaluation determined that the installation of this temporary modification did not increase the probability of occurrence of an accident previously evaluated in the Final Safety Analysis Report (FSAR) because the radiation monitors are used to mitigate the consequences of a refueling accident and a loss of coolant accident (LOCA). It did not degrade or prevent actions described or presumed in the refueling accident or LOCA.

This safety evaluation determined that the installation of this temporary flow indicator and manual operation of the flow control valve did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-003 **Rev. 0** **INSTALLATION OF DUCT AND REMOVAL OF CO2 OPERATOR AT ELECTRICAL BAY FIRE DAMPERS 67CD-3 AND 67CD-4**

Modification: M1-95-007

The purpose of this safety evaluation was to evaluate Modification M1-95-007 which added a duct to the Turbine Building Ventilation System Fire/CO2 Dampers 67CD-3 and 4 located in the East and West Electric Bays, respectively, and removed the automatic closure feature on the dampers during CO2 initiation. This modification was necessary to prevent the overpressurization of the Electric Bays during a CO2 discharge. An analysis of the Electric Bays had determined that inadequate venting capability existed during discharge of the CO2 Suppression System.

The duct that was installed at the CO2 damper is not located near any safety-related equipment. Analysis of the CO2 System capability in the Electric Bays indicated that the modification will return the CO2 Suppression System operation to its original design and ensure the integrity of the fire barrier thereby ensuring the protection of the safety-related equipment located in the Electric Bays.

The Final Safety Analysis Report (FSAR) Section 9.8.3.2, "Carbon Dioxide Systems" was revised to address the design changes. No change to Technical Specifications was required as a result of this modification.

This safety evaluation determined that this modification did not constitute any unreviewed safety questions pursuant to 10 CFR 50.59.

JAF-SE-95-004 Rev. 0 RELAY ROOM ENCLOSURE INTEGRITY TEST

Modification: N/A

This safety evaluation documents the acceptability of the performance of a tracer gas and door fan test in the Relay Room. This test allowed for the collection of test data that was required to support the subsequent evaluation and verification of the design basis and system performance for the CO2 Fire Suppression System in this room. This enclosure integrity test was an acceptable alternative to full discharge testing of the CO2 Suppression System. The implementation of this test did not produce any permanent, physical or functional changes to the CO2 Suppression System in the Relay Room.

This enclosure integrity test did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety evaluated previously in the Final Safety Analysis Report (FSAR). This test did not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the FSAR.

This safety evaluation provided the bases for the determination that the performance of the enclosure integrity test did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-005 Rev.1 TEMPORARY OPERATING PROCEDURE TOP-203

Modification: N/A

The purpose of this safety evaluation was to address the acceptability of a new Temporary Operating Procedure (TOP-203) to allow shutdown of the Relay Room Ventilation System to perform Special Test Procedure STP-76AU, "Relay Room Enclosure Integrity Test".

The scope of this evaluation was to address the consequences of shutting down the Relay Room Ventilation System in accordance with TOP-203 to allow specific test activities to be performed. Revision 1 to this safety evaluation was issued to include in the evaluation limited work activities required by Modification F1-92-377.

Modification F1-92-377 required the installation of Electro-Thermal Links for Relay Room Ventilation System fire dampers 70FD-1 and 2 and the installation/removal of electrical jumpers in junction box JBCRV-33.

TOP-203 provided direction to Operations for actions necessary to shutdown and restore the system, delineate the appropriate Technical Specification requirements, and describe the necessary compensatory actions that had to be taken during system shutdown to ensure that the required system safety functions were maintained.

This evaluation determined that the performance of TOP-203 did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety evaluated previously in the Final Safety Analysis Report (FSAR) because room air temperature limits were not exceeded. This TOP did not create the possibility for an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR because this change only allowed system shutdown for short durations to allow testing and limited work activity.

This safety evaluation concluded that the performance of TOP-203 did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-007 Rev. 0 ADDITION OF AUTOMATIC OPEN/CLOSE FUNCTION TO 72AOD-137

Modification: M1-95-010

The purpose of this modification was to add automatic open/close capability to air operated damper 72AOD-137 during actuation of the CO2 Fire Protection System in the North Cable Run Room. This feature is required to prevent over-pressurization of the North Cable Run Room during a CO2 discharge. An analysis of the subject space had determined that inadequate venting capability existed and thus corrective measures were necessary. As a result of this modification 72AOD-137 opens/closes automatically upon CO2 activation/de-activation to provide the necessary vent path.

The implementation of this modification did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety. The modification does not create the possibility of an accident or malfunction of a different type previously evaluated in the Final Safety Analysis Report (FSAR) and does not reduce the margin of safety of the Fire Protection System.

This safety evaluation concluded that this modification does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-010 **Rev. 0** **RPV LEVEL CONTROL FOR REFERENCE COLUMN 2A PIPING REPLACEMENT**

Modification: **N/A**

The purpose of this evaluation was to review the effects of Temporary Operating Procedure TOP-205 on plant safety. TOP-205 was required to control the evolution of plant operations to allow 2A and 3A reference column reactor water level and pressure instruments to be valved out-of-service while meeting the Action Statements of the applicable sections of the Technical Specifications. The modification replaced a section of 1 ½ inch piping to the 2A condensing chamber.

TOP-205 removed both the 2A and 3A reference columns from service for the installation of Modification F1-93-075. This required the pressure and level instruments associated with 2A and 3A reference column to be removed from service. The significance of taking the "A" side reference column out of service was that this removes half of the initiating logic for the Emergency Core Cooling System (ECCS) initiation, the Primary Containment Isolation System (PCIS), the Reactor Protection System (RPS) and Secondary Containment.

The plant remained in the Cold Shutdown condition for refuel during the pipe replacement evolution.

The safety evaluation determined that the probability of occurrence of an accident or equipment malfunction was not increased because the systems required for safe operation were not affected and provisions/controls were taken to prohibit certain activities. The consequences of an accident or equipment malfunction were not increased because the credible scenarios (loss of inventory, increase in reactor temperature) were bounded by manual ECCS operation if required. Accidents or equipment malfunctions of a different type then those described in the Final Safety Analysis Report (FSAR) were not created because the TOP involved the operation of existing available systems within the requirements of the Technical Specifications. The margin of safety as defined in Technical Specifications was not reduced because provisions/controls for suspending certain activities were included in TOP-205 to ensure safety system availability as required by the Action Statements in Technical Specifications.

This safety evaluation concluded the effects of performing TOP-205 did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-011 **Rev. 0** **FSAR CHANGE TO INTERIOR FIRE HOSE STATIONS**

Modification: **N/A**

This safety evaluation was the basis for determining whether any safety concerns existed associated with a change to the Final Safety Analysis Report (FSAR) Chapter 9.8, "Fire Protection Systems". The change involved the limitation of the length of fire hose on interior fire hose stations.

FSAR Section 9.8.3.1.3 called for fire hose stations in the plant to be equipped with 50 foot and 75 foot lengths of fire hose. On the contrary, fire hose stations in the plant were equipped with 100 feet of fire hose. The FSAR was revised to reflect this condition. A hose length study verified that with the hose lengths that are installed on the hose stations (100 feet), all areas containing safety-related equipment can be reached.

This revision did not change the function, performance, or integrity of boundaries that form or support the primary protective barriers (fuel cladding, primary pressure or containment) on which the consequences of previously identified accidents were based. This change did not contribute to any of the failure modes analyzed in the FSAR nor did it affect a system such that a new failure mode would occur. This change did not degrade or prevent any action assumed to occur in the FSAR that would mitigate the consequences of a malfunction of safety-related structures, systems or components.

This safety evaluation concluded that the revision to the FSAR, Section 9.8.3.1.3 did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-012 Rev.0 FSAR CHANGE TO RESTRICTED AREA TURBINE BUILDING FIRE PROTECTION SYSTEM DESCRIPTION

Modification: N/A

This safety evaluation was the basis for determining whether any safety concerns existed associated with a change to the Final Safety Analysis Report (FSAR) Chapter 9.8, "Fire Protection Systems". The change involved the temperature setting of the heat detectors for the Fire Protection System in the Restricted Area of the Turbine Building.

FSAR Section 9.8.3.1.4 included that where automatic sprinklers are installed in the Restricted Areas of the Turbine Building thermostats were set at a lower temperature than the sprinkler to indicate presence of fire before sprinkler actuation and to indicate when the fire was extinguished. The FSAR was changed to explain that the Heat Detection System may be used to determine fire extinguishment and that the Fire Detection System is basically a backup system to the automatic sprinkler system.

The setpoint for the thermostats were either identical or they were so close to sprinkler fusible temperature that a pre-alarm by the thermostats was inconsequential to initiate any action prior to sprinkler actuation.

This temperature setting revision did not change the function, performance, or integrity of boundaries that form or support the primary protective barriers (fuel cladding, primary pressure or containment) on which the consequences of a previously identified accident are based. This activity did not contribute to any of the failure modes analyzed in the FSAR nor did it effect a system such that a new failure mode would occur.

This safety evaluation concluded that the FSAR change, as described, did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-013 **Rev. 0** **INSTALLATION OF BLANK FLANGE IN PLACE OF RUPTURE DISC PARALLEL TO 33RV-101**

Modification: **N/A**

The purpose of this safety evaluation was to evaluate the acceptability of installing a blank flange in place of the rupture disc which along with Condensate System relief valve 33RV-101 provides overpressure protection to the Mixed Resin Storage Tank 33TK-15.

33TK-15 is configured with two pressure protective devices connected in parallel. The principal means of overpressurization is 33RV-101. This relief valve has a setpoint of 75 pounds per square inch gauge (psig). The secondary means of overpressurization protection was a rupture disc with a burst pressure (setpoint) of 100 psig. It was necessary to remove the rupture disc and replace it with a blank flange due to piping misalignments which caused failures of the fragile disc.

This safety evaluation concluded that the replacement of the rupture disc with a blank flange was acceptable. Relief valve 33RV-101 had sufficient capacity to provide overpressurization protection for tank 33TK-15. The 75 psig set for 33RV-101 corresponds to the allowable working design pressure for 33TK-15. Any concerns relative to the plugging of the relief valve operator were eliminated by Modification D1-95-033 which replaced the valve operator and improved valve reliability in the event resin was introduced into the relief valve operator assembly.

The activities identified by this safety evaluation did not increase the probability of occurrence of an accident evaluated in the Final Safety Analysis Report (FSAR) nor create the possibility of an accident of a different type than any previously evaluated in the FSAR. The margin of safety as defined in any Technical Specifications was not reduced because the Condensate Demineralizer System is not discussed in the Technical Specifications.

This safety evaluation concluded that the replacement of the rupture disc with a blank flange did not involve an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-014 Rev. 2 STEAM LINE BREAK EFFECT ON ESSENTIAL ELECTRICAL EQUIPMENT IN THE EAST AND WEST ELECTRICAL BAY AND THE EAST AND WEST CABLE TUNNEL

Modification: N/A

As part of a review of Nuclear Regulatory Commission (NRC) Information Notice 92-52, "Barriers and Seals Between Mild and Harsh Environments", ventilation louvers containing fire dampers in the walls of the Electric Bays were found open to the Turbine Building. This potential leakage path had not been considered in the original analysis of High Energy Line Breaks (HELB) in the Turbine Building. The Electric Bays were considered a "mild environment" for the purpose of internal equipment qualification. Also, safety-related control panels for the Electric Bays and Cable Tunnel Ventilation System located inside the Turbine Building had not been evaluated. Modification D1-94-239 moved the remaining Electric Bay electrical ventilation controls to inside the East Electric Bay. Cable tunnel ventilation controls were not relocated.

A re-analysis of the Turbine Building HELB calculation and the effect of potential loss of Cable Tunnel ventilation for a 48 hour period was performed.

It was concluded that with the ventilation louvers and duct work to the Turbine Building open to a postulated HELB in the Turbine Building that a harsh environment did not exist in the Electric Bays. In addition, based on analyses, the loss of ventilation in the Cable Tunnels can be tolerated for 24 hours. Cold shutdown can be achieved after a Main Steam Line Break in 24 hours. Operator actions are required to assure proper cooling for the Emergency Diesel Generator Switchgear Rooms.

This evaluation concluded that the effects on essential electrical equipment in the East and West Electric Bays and in the East and West Cable Tunnels by the most severe steam line break in the Turbine Building did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-015 Rev.0 EVALUATION OF LOST PARTS IN THE REACTOR CAVITY DURING RELOAD 11 REFUELING OUTAGE

Modification: N/A

This safety evaluation was the basis for determining that parts unaccounted for at the completion of work in the reactor cavity at the end of the Reload 11 refueling outage did not create an unreviewed safety question.

A number of small items were not accounted for as having been located and removed from the reactor cavity following Reload 11 refueling, therefore, they were assumed to have entered the Reactor Pressure Vessel (RPV). The largest durable item in this identified population was a stainless steel 3/8 inch x 16 socket head cap screw, 1 to 1-1/4 inches long.

This safety evaluation considered the following potential effects of the identified objects (as applicable):

- Interference with control rod operation,
- Fuel bundle flow blockage and subsequent fuel cladding overheat and damage,
- Corrosion or adverse chemical reaction with other reactor materials,
- Interference with Reactor Water Cleanup (RWCU) System line isolation valves,
- Fuel damage due to fretting wear,
- Impact on Recirculation System performance,
- Flow blockage of bottom head drain line,
- Damage to RWCU System pumps.

It was demonstrated that the lost parts either would cause no effect (e.g., control rod interference, RWCU isolation valve interference) or the effects were bounded by previous analyses.

This safety evaluation concluded that the presence of the number of identified small items in the RPV did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-018 Rev.0 DRYWELL ENTRIES DURING STARTUP AND SHUTDOWN

Modification: N/A

This nuclear safety evaluation was the basis for determining whether or not personnel entry into the Drywell during plant startup and shutdown presented an unreviewed safety question. This document evaluated the acceptability of allowing Drywell entries during power operation with reactor conditions of:

- Less than or equal to 10 percent rated thermal power,
- Reactor mode switch not in Run position,
- A controlled reactor power level with no planned evolutions which could result in plant power changes.

The Final Safety Analysis Report (FSAR) Section 5.2.3 restricted Drywell entry during power operation. This safety evaluation provided the basis to change the FSAR to allow Drywell entry provided the above conditions were met. Controls were established through procedure implementation to ensure and protect the health and safety of workers entering the Drywell environment.

This evaluation concluded that worker entry into the Drywell at power would not increase the probability of occurrence of accidents or malfunction of equipment important to safety because the presence of personnel in the Drywell does not alter the design or operation of plant systems. The consequences of accidents evaluated in the FSAR would not be increased because Primary Containment integrity will be maintained at all times in accordance with Technical Specifications.

This safety evaluation concluded that entry of personnel into the Drywell during the plant conditions specified above did not constitute an unresolved safety question pursuant to 10 CFR 50.59.

JAF-SE-95-019 Rev. 2 LEAK REPAIR OF 31MOV-RSSV-1

Modification: N/A

The purpose of this safety evaluation was to evaluate the acceptability of applying a fifth application of a temporary leak repair compound to the Moisture Separator Reheater Steam Supply Isolation Valve 31MOV-RSSV-1 as required in Maintenance Procedure MDSO-3, "Temporary Leak Repair". The valve had exhibited a persistent steam leak since startup from the 1994-1995 Refuel Outage. Leakage had been reduced to a small wisp from the valve packing area after four applications of the leak repair compound.

The repair activity included injecting a chemical compound (sealant) into a cavity in the upper bonnet area of the valve to eliminate steam leakage out of the valve pressure boundary to the general area.

Valve 31MOV-RSSV-1 is a normally open valve that provides main steam to the second stage moisture separator reheater. This valve serves no safety-related function.

The effects of the leak repair included the additional valve weight on adjacent pipe supports, structural integrity of the valve, the chemical effects on the Main Steam System, and the effects on valve operability were reviewed in this evaluation The evaluation concluded that negligible impact existed in these areas. Based on the above analysis it was determined that the fifth application of leak repair in the over bonnet area of valve was acceptable.

This safety evaluation concluded that the repair activity to 31MOV-RSSV-1 did not involve an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-020 Rev. 0 INTERNAL INSPECTION OF THE CONDENSATE STORAGE TANKS, 33TK-12A/B

Modification: N/A

The purpose of this evaluation was to evaluate the acceptability of utilizing an underwater submarine unit with an attached video camera to perform an internal inspection of the Condensate System Storage Tanks 33TK-12A and B.

The following Final Safety Analysis Report (FSAR) sections were reviewed relative to performing an internal inspection of the Condensate Storage Tanks (CSTs):

(Systems taking suction from the CSTs)

- Control Rod Drive Mechanisms
- Standby Liquid Control System
- Emergency Core Cooling Systems
- Condensate Storage System
- Reactor Core Isolation Cooling System
- High Pressure Coolant Injection System

The review concluded that the CSTs would remain operable per their design function during the internal inspections. Additionally, minimum CST water level requirements, as specified in Technical Specifications, would be maintained at all times.

The evaluation concluded that the internal inspection of the CSTs would not increase the probability of occurrence of an accident evaluated in the FSAR nor would the margin of safety be reduced as defined in the Technical Specification because at no time will the CSTs be inoperable.

This safety evaluation concluded that the inspection evolutions required for the CSTs did not involve an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-021 **Rev. 0** **UPDATE OF FSAR SECTION 13.3, TRAINING OF PERSONNEL**

Modification: **N/A**

The purpose of this safety evaluation was to review the updates required to the Final Safety Analysis Report (FSAR), Section 13.3, "Training of Personnel" for compliance with Code of Federal Regulations 10 CFR 50.120, "Training and Qualification of Nuclear Power Plant Personnel" and 10 CFR 55, "Operators Licenses".

FSAR Section 13.3 identifies the requirements for General Employee Training, Licensed Operator qualification and retraining requirements, non-licensed operator training, technicians, maintenance personnel, and professional and supervisory personnel. This section contains detailed requirements for training, qualification, and evaluation. The FSAR update reflects compliance to the 10 CFR 50.120 training regulation and to the 10 CFR 55 Operator License regulation.

This evaluation concluded that the training programs are in full compliance with the regulations as accredited by the Institute of Nuclear Power Operation (INPO) National Academy for Nuclear Training. The training programs provide both initial and continued training to establish and maintain the overall proficiency of the staff and to provide qualified personnel to operate and maintain the plant in a safe manner in normal, abnormal, and emergency conditions.

This safety evaluation concluded that the update to the FSAR does not involve an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-027 Rev. 1 LEAK REPAIR OF 31AOV-RSLLV2

Modification: N/A

The purpose of this safety evaluation was to evaluate the acceptability of applying a fifth application of a temporary leak repair compound to the Extraction Steam System "B" Moisture Separator Second Stage Reheater Low Load Isolation Valve (31AOV-RSLLV2). The valve had developed a recurring body to bonnet steam leak.

The repair activity included injecting a chemical compound (sealant) into the bolt ring of the valve body to bonnet area to eliminate steam leakage out of the valve pressure boundary.

31AOV-RSLLV2 is a normally open valve at full power that provides main steam to the Second Stage Moisture Separator Reheater. This valve can operate automatically to control low power reheat or be controlled remotely from the local panel in the Turbine Building during startup and shutdown of the Moisture Separator Reheater Drain System.

The effects of the leak repair including the additional valve weight on pipe supports, structural integrity of the valve, the chemical effects on the Main Steam System, and the effects on valve operability were reviewed in the evaluation. The evaluation concluded that negligible impact existed in these areas. Based on the above analysis it was concluded that the fifth application of leak repair in the body to bonnet area of 31AOV-RSLLV2 was acceptable.

The evolution did not increase the probability of occurrence of an accident evaluated in the Final Safety Analysis Report (FSAR) nor did it increase the consequences of an accident evaluated in the FSAR. This valve does not perform a safety function and is located in a non-safety-related portion of the Main Steam System.

This safety evaluation concluded that the fifth application of sealant to 31AOV-RSLLV2 did not involve an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-028 Rev.0 RE-ROUTE OF 02-2RV-42A, B LEAKAGE TO EQUIPMENT DRAINS

Modification: N/A

The purpose of this evaluation was to determine the acceptability of temporarily re-routing the discharge from the Reactor Water Recirculating System pump mini-purge relief valves, 02-2RV-42A, B, from the floor drain subsystem of radwaste to the sample sink, 95SP-1. 95SP-1 drains to the equipment drain subsystem of radwaste. This temporary deviation was needed because of leaks in the mini-purge relief valves. The leakage was caused by system resistance changes bringing line pressure too close to the relief valve setpoint.

The temporary modification required the use of a heavy duty hose routed away from safety-related equipment in the area. The hose was connected to the mini-purge relief valve drain line between the Reactor Building Closed Loop Cooling System heat exchangers and routed to the sample sink.

This evaluation determined that the activities described above did not increase the probability of occurrence of an accident evaluated in the Final Safety Analysis Report (FSAR) because no credible failure mode of this temporary modification could have initiated any analyzed or unanalyzed plant transient or accident. The mini-purge relief valve discharge to the radwaste system has no design basis safety function and is not credited in any plant transient or accident analysis.

This safety evaluation concluded that the temporary modification described above did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-030 Rev. 0 EVALUATION OF CONTROL ROD 22-39 FAST WITHDRAWAL VELOCITY

Modification: N/A

The purpose of this safety evaluation was to evaluate whether plant operation with Control Rod Drive (CRD) 22-39 withdrawing at a faster than normal rate posed an unreviewed safety question.

During CRD surveillance testing full withdrawal of CRD 22-39 (position 00 to 48) occurred faster than the specified "as-found" band of 39 to 57 seconds. Attempts to adjust the CRD withdrawal time to within the specified "as-left" range of 45 to 51 seconds were unsuccessful. The final rod withdrawal time ranged from 32 to 39 seconds. The last recorded time was 39 seconds which corresponds to a withdrawal speed of 3.7 inches per second. Insert drive times were acceptable.

This evaluation concluded that CRD withdrawal speeds of up to 5 inches per second were acceptable. This corresponds to a withdrawal time of 28.8 seconds (position 00 to 48). As a result, withdrawal of CRD 22-39 in any time greater than 28.8 seconds does not adversely impact the Final Safety Analysis Report (FSAR) abnormal transient or accident analyses.

No changes to the FSAR were required as a result of the increased withdrawal velocity of rod 22-39. The rod withdrawal speed did not affect the SCRAM function of the rod. The consequences of an accident related to rapid withdrawal of a control rod are bounded by the Rod Drop Accident Analysis in the FSAR.

This safety evaluation concluded that plant operation with Control Rod Drive 22-39 withdrawing at a faster than normal rate did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-031 Rev. 1 INSTALL ARC SUPPRESSION CAPACITORS ACROSS UNLATCH, ROD WITHDRAW, AND SETTLE RELAYS (03A-K14, 03A-K15, AND 03A-K19)

Modification: N/A

The purpose of this safety evaluation was to evaluate the acceptability of adding .01 microfarad capacitors rated for 2000 volts direct current (DC) across the coils of "Unlatch, Rod Withdraw and Settle" relays 03A-K14, 03A-K15, and 03A-K19. These capacitors were necessary to correct a timing problem which resulted in shorting the duration of the rod withdrawal output signal in the Reactor Manual Control System (RMCS). The three capacitors were added across the coils circuits of 03A-K14, 03A-K15 and 03A-K19. These capacitors absorbed transient energy which had been demonstrated to corrupt rod withdrawal count sequence of the RMCS solid state timer.

All relevant sections of the Final Safety Analysis Report (FSAR) were reviewed with no impact on the design bases. The parameters impacted by this change are not discussed in the Technical Specifications.

The evaluation determined that the activities described above did not increase the probability of occurrence of an accident evaluated in the FSAR because the design basis safety function of the RMCS was not impacted. The RMCS circuitry remained independent of the scram circuitry as described in the FSAR. The installation did not increase the consequences of an accident evaluated previously in the FSAR because The RMCS serves no accident mitigation function. No new failure modes were introduced and no design basis assumptions were challenged by this change.

This safety evaluation concluded that the changes described above did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-032 **Rev. 0** **USE OF HI-HI RADIATION ISOLATION OVERRIDE SWITCHES DURING 27RM-104A/B MAINTENANCE**

Modification: **N/A**

This safety evaluation was written to allow the operation of the Primary Containment Purge (PCP) System using the override switches when the system is isolated due to one Containment High Range Radiation Monitor (27RM-104A/B) being out of service for maintenance/testing. The removal of either of these monitors results in a FAIL/HI-HI RADIATION isolation of either the inboard ("B" monitor) or outboard ("A" monitor) vent and purge valves. Removal of both monitors results in an isolation of both inboard and outboard valves. Use of the key-lock override switches was previously only allowed in the Emergency Operating Support Procedures.

Operation of the PCP System with one radiation monitor out of service is done by using the isolation logic override switch to manually reset the isolation for the time needed. This is acceptable and is consistent with the design of the system. As the affected valves are normally closed, no actual valve movement occurs as a result of the isolation. Upon completion of vent/purge operations, the key-lock override switch is returned to NORMAL position to again seal in the isolation signal. This is acceptable because:

- The valves which receive isolation signals are normally closed,
- The function of the operable radiation monitor and primary containment isolation instrumentation to automatically isolate associated Containment Air Dilution (CAD) System valves is unaffected,
- Technical Specification Table 3.2-8, Accident Monitoring Instrumentation, allows for one radiation monitoring channel to be out of service,
- Technical Specification Bases 3.7.D, Primary Containment Isolation Valves, allows for valves to be opened periodically for venting the Torus to maintain the required Torus to Drywell differential pressure.

This evaluation concluded that the margin of safety was not reduced as defined in Technical Specifications. The bases for the containment radiation monitoring and containment isolation do not prohibit the proposed activity. The minimum number of instrument channels are maintained. Containment isolation requirements are unaffected. As such, the use of these override switches for the stated purpose does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-033 Rev. 0 VACUUM FILTER REMOVAL FROM THE SPENT FUEL POOL

Modification: N/A

This safety evaluation was the basis for determining the suitability of using the On-Site-Storage-Container (OSSC) for transportation of the underwater vacuum filters generated as a result of outage activities from the Spent Fuel Pool (SFP) and for storage of these filters in the Interim Waste Storage Facility (IWSF).

The OSSC is a concrete container normally used for the storage of radioactive waste. Lifting the OSSC to the Refuel Floor and using it for transportation of radioactive waste from the SFP was the first application of this container for this purpose at the FitzPatrick Plant as well as the first in the industry. As a result, the OSSC was load-tested outside the plant prior to bringing it to the Refuel Floor. The entire handling of the empty and loaded OSSC was performed in accordance with NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants". Other potential concerns in this cleanup effort were hot particle contamination, extremity overexposure, increased radiation levels on the Refuel Floor and shipment of highly irradiated materials in a concrete cask. Lessons learned from similar tasks in the past were incorporated into the As-Low-As-Reasonably-Achieveable (ALARA) reviews and as a result the radiation levels of the entire operation were extremely low and controlled.

The IWSF was designed and evaluated for the storage of waste in containers like the OSSC in a previous safety evaluation and found acceptable for the purpose. This safety evaluation concluded that the removal of vacuum filters in the OSSC and storing the OSSC in the IWSF did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-034 Rev.0 EVALUATION OF REMOVAL OF VARIOUS CONTAINMENT ISOLATION VALVES FROM AP-01.04

Modification: N/A

This safety evaluation was written to support a revision to Administrative Procedure AP-01.04, "Tech. Spec. Related Requirements, Lists and Tables". Specifically, several valves were included in AP-01.04, Attachment 1 (Primary Containment Isolation Valves) although the valves did not perform a containment isolation function.

The scope of the procedure revision involved the deletion of Reactor Building Closed Loop Cooling System and Emergency Service Water System manual and check valves from AP-01.04, Attachment 1. The subject manual and check valves are installed in process lines that also contain remotely operated air operated valves (AOVs). The AOVs provide the containment isolation function and meet 10 CFR 50, Appendix A, General Design Criteria 57, "Closed System Isolation Valves". Type C local leak rate testing was performed on both the manual and check valves in addition to the AOVs. This revision eliminated the Type C leak rate test requirements for the manual and check valves and removed the valves from the In-Service Testing (IST) Program.

Removal of these valves from AP-01.04 did not conflict with the Final Safety Analysis Report (FSAR) requirements for containment isolation. These requirements are entirely met by other valves in the applicable penetrations. The revision did not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the FSAR. The containment will still provide a barrier as described in the FSAR. All assumptions made and design criteria concerning the design of the lines penetrating the containment remain valid.

This safety evaluation concluded that the deletion of certain containment isolation valves from AP-01.04, Attachment 1, did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-036 Rev. 0 JUSTIFICATION TO SECURE FUEL POOL COOLING AS DIRECTED BY TOP-212, OPERATION OF THE FUEL POOL COOLING SYSTEM DURING HYDROLASING OF FUEL POOL COOLING LINES

Modification: N/A

The purpose of this evaluation was to address the acceptability of securing the Spent Fuel Pool Cooling System and partially draining the Skimmer Surge Tanks to perform Temporary Operating Procedure TOP-212, "Operation of the Fuel Pool Cooling System During Hydrolasing of Fuel Pool Cooling Lines".

The performance of TOP-212 deviated from the Final Safety Analysis Report (FSAR) in that the FSAR states that the Fuel Pool Cooling System is normally in operation while fuel is stored in the Spent Fuel Pool. This safety evaluation was completed to evaluate taking the Spent Fuel Pool Cooling System temporarily out-of-service during fuel storage.

The performance of TOP-212 deviates from the Final Safety Analysis Report (FSAR) relative only to the descriptive section and does not impact system safety. A review of potential failure modes of the Fuel Pool Cooling System (i.e.; maintaining water level limits, maintaining an operable alternate supply water source, monitoring of fuel pool water temperature) and compensatory actions provided in TOP-212 determined that the performance of TOP-212 did not increase the probability of occurrence of an accident described in the FSAR.

Operation of the Spent Fuel Pool Cooling System is not credited with mitigating the consequences of any accident described in the FSAR. The performance of TOP-212 did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the FSAR. The liner integrity was not compromised because the water temperature and level were monitored hourly. The performance of TOP-212 did not reduce the margin of safety as defined in the basis for Technical Specifications because the performance of TOP-212 did not change any safety limit, limiting safety system setting, limiting conditions for operation or any design parameter of the Spent Fuel Pool Cooling System.

This safety evaluation concluded that the performance of TOP-212 did not constitute an reviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-037 **Rev. 0** **ADDITION OF PORC NON-QUORUM VOTING MEMBERS**

Modification: **N/A**

This safety evaluation assessed the significance and potential impact on the Technical Specifications of adding the Licensing Manager and the Quality Control Manager to the Plant Operating Review Committee (PORC) as non-quorum voting members. The purpose was to provide additional expertise for PORC. Senior management determined that the benefit to the review process by these two positions was such that their immediate inclusion on PORC was warranted. Administrative controls were established assuring compliance with Technical Specification requirements regarding PORC. A subsequent Technical Specification Amendment established these as permanent, full PORC members.

It was concluded that the proposed changes were administrative in nature and that they were consistent with the stated function of PORC to "advise the Resident Manager on all matters related to nuclear safety and all matters which could adversely change the plant's environmental impact". It was concluded that the changes did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-039 **Rev. 0** **STORAGE OF STEAM LINE PLUGS AND CATTLE CHUTE IN THE INTERNAL STORAGE PIT**

Modification: **N/A**

The condition of the floor paint on the Refuel Floor required measures to be taken to prepare the floor for repainting. To perform this task it was advisable to remove highly contaminated refueling equipment like the Portable Radiation Shield commonly known as the "cattle chute" and the steam plugs from the floor during the duration of the painting or preferably on a permanent basis. This safety evaluation reviewed the safety concerns for permanent relocation of these items from the Refuel Floor to the Internal Storage Pit (ISP) during non-outage periods. Storage of these items on the ISP Floor for the duration of the entire non-outage period reduces the dose rate on the Refuel Floor and also saves approximately six hours of critical outage time for wrapping and relocating them to the Refuel Floor.

The concerns associated with this storage analysis were floor loading (structural stability), heavy loads handling, and rigging. Handling of loads were performed to the requirements of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants". Floor loads were evaluated per the design criteria of plant structures and components. The units were found to be stable and within the allowable loads for the ISP. The evaluation concluded that the movement of the "cattle chute" and the steam plugs with the reactor building crane and relocating them in the ISP by methods and requirements outlined in the safety evaluation did not involve significant hazards.

This safety evaluation concluded that activities cited did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-042 Rev. 0 CHEMICAL TREATMENT OF CRESCENT AREA UNIT COOLERS

Modification: N/A

In 1990 and 1991 the small bore Service Water System (SWS) and Emergency Service Water (ESW) System piping were found to have moderate to heavy Microbiologically Induced Corrosion (MIC) in the form of nodules. This resulted in a 10 to 50 percent occlusion of the piping. To remove these deposits from the piping chemical treatment of all affected areas began in August 1992. Evaluations of the treatment effectiveness in improving the internal piping condition have concluded that the Crescent Area Unit Coolers were not receiving an adequate concentration of chemicals. To address this issue, off-line chemical soak treatments were performed on individual Crescent Area Unit Coolers. Two chemicals in a dilute solution with water were recirculated through the out-of-service unit cooler. At the completion of the treatment period the solution was discharged to Lake Ontario through the normal Service Water System discharge path and the unit cooler was then flushed. These chemicals result in dissolving and then dispersing the corrosion nodules into the flow stream. This process results in improving unit cooler ESW System flow rates and ensuring that each heat exchanger is capable of removing its design accident heat load. The chemicals do not have detrimental effects on system materials and have been approved for use and discharge to the lake by the New York State Department of Environmental Conservation.

The chemicals are only applied to an out of service unit cooler. The removal of one (1) unit cooler from service in a given Crescent Area does not affect plant safety because only four (4) of the five (5) unit coolers are required to remove the total accident heat load.

This safety evaluation concluded that the chemical treatment activity to the Crescent Area Unit Coolers did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-043 Rev. 1 EVALUATION OF ELEVATED LAKE TEMPERATURE

Modification: N/A

This safety evaluation was performed to support plant operation with an 82.4 degree Fahrenheit Service Water temperature, specifically, for the impact of the 82.4 degrees Fahrenheit on the performance of the Residual Heat Removal Service Water (RHRSW) and the Emergency Service Water (ESW) Systems. This evaluation was based on an examination of the impact of the higher service water temperature on Safety Relief Valve (SRV) operation, Loss of Coolant Accident (LOCA) related containment responses, Emergency Core Cooling System (ECCS) performance during design basis events, the operability of the RHRSW System pumps, and ECCS performance for degraded events. It was concluded that the 82.4 degree Fahrenheit service water temperature did not have an adverse affect on any of the above items. Evaluations were performed to determine that all applicable regulatory requirements were satisfied. There was no effect on system and component functional design and safety bases as defined in the Final Safety Analysis Report (FSAR). Plant Technical Specifications were reviewed to assess the effects on applicable Limiting Conditions for Operation, Limiting Safety System Settings, Safety Limits, and reactor thermal parameters and concluded that the 82.4 degree Fahrenheit temperature for the service water did not reduce the margin of safety as defined in the bases for the Technical Specifications. No changes in the Technical Specifications were required due to operation with lake water temperature of 82.4 degrees Fahrenheit. The 82.4 degree Fahrenheit temperature analysis is valid until a more limiting lake water temperature value is determined based on future performance of Surveillance Tests ST-19C, G, and H (Crescent Area, Electric Bay, and Cable Tunnel Ventilation Unit Cooler Performance Test With ESW Flow). At that time, this evaluation would need to be revised. The design basis lake temperature limit will be restored to 82 degrees Fahrenheit pending completion of an Engineering effort to determine the maximum attainable lake water temperature limit which is expected to be completed by August 1996.

This safety evaluation concluded that the plant could safely operate with a service water temperature of 82.4 degrees Fahrenheit and that this operation did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-044 **Rev. 0** **JUSTIFICATION TO SECURE FUEL POOL COOLING TO SUPPORT MAINTENANCE ACTIVITIES DURING CYCLE 12**

Modification: **N/A**

The purpose of this safety evaluation was to address the acceptability of securing the Spent Fuel Pool Cooling System to perform maintenance activities which require the system to be shut down.

The Final Safety Analysis Report (FSAR) Section 9.4.4 states that the system is normally in operation while fuel is stored in the spent fuel pool. Operating Procedure OP-30 "Fuel Pool Cooling and Cleanup System" also contains guidance to operators which states each time the Spent Fuel Pool Cooling System is shut down a safety evaluation is required. This safety evaluation was used to change OP-30 to reference this safety evaluation for justification during each system shutdown during this cycle (cycle 12).

The impact of temporarily securing the Spent Fuel Pool Cooling System was reviewed relative to applicable section of the FSAR and Technical Specifications. Safety Limits including Spent Fuel Pool temperature and water level were reviewed. It was concluded that adequate instruction and compensatory action were in place to assure that required system safety functions were maintained.

The review determined that temporarily securing the Spent Fuel Pool Cooling System had no adverse impact on normal plant operation, safe plant shutdown or accident mitigation. Operation of the Spent Fuel Pool Cooling System is not credited in any abnormal operational transient or design basis accident described in the FSAR.

This safety evaluation concluded that the activity described above did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-046 Rev. 0 AP-01.04, ATTACHMENT 2, FIRE PROTECTION CHANGES

Modification: N/A

The purpose of this safety evaluation was to evaluate any safety concerns associated with changes (draft revision 9) to Administrative Procedure AP-01.04, "Tech. Spec. Related Requirements, Lists and Tables", Attachment 2, Fire Protection Requirements.

The changes associated with AP-01.04 Attachment 2 involved an increase in the surveillance test interval for the functional testing of fire detectors, the removal of fire protection impairment reporting requirements to the Nuclear Regulatory Commission, editorial changes that provide clarification and definition, incorporation of two Fire Protection Technical Specification interpretations, the addition of 10 CFR 50 Appendix R boundary-related water spray curtains, associated Heat Detection Systems and area Smoke Detection Systems, the isolation of CO2 Fire Suppression Systems for personnel protection, a change in the compensatory measures (i.e. fire watch requirements) associated with CO2 Fire Suppression System and Water Spray/Sprinkler Systems, and a change in the test frequency associated with fire hose stations in high radiation areas.

This safety evaluation determined that the proposed (Revision 9) changes to AP-01.04 Attachment 2 did not change the state or function of the Fire Protection System as described in Revision 8 to AP-01.04 or Chapter 9 of the Final Safety Analysis Report (FSAR). The changes did not alter any inputs or assumptions for the probabilities of accidents previously evaluated. The margin of safety as defined in the basis of the Technical Specifications has not been reduced because this activity does not affect the Technical Specifications or the safety analysis of the plant as described in the FSAR.

This safety evaluation concluded that this revision to Administrative Procedure AP-01.04 did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-050 **Rev. 0** **MANAGEMENT POSITION TITLE CHANGES: CHANGES TO THE MANAGEMENT POSITION TITLES OF VICE PRESIDENT REGULATORY AFFAIRS AND SPECIAL PROJECTS, AND VICE PRESIDENT-NUCLEAR ENGINEERING**

Modification: N/A

The purpose of this safety evaluation was to evaluate the safety significance of the New York Sate Power Authority's organization changes which changed the position of "Vice President Regulatory Affairs and Special Projects" to "Director of Regulatory Affairs and Special Projects" and changed the position of 'Vice President-Nuclear Engineering" to "Director-Nuclear Engineering". Both were changes in title only with no changes in incumbency, responsibility, reporting relationships, or qualifications.

This evaluation concluded that the changes were administrative in nature with no effect on nuclear safety and that the changes did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-051 **Rev. 0** **RWCU TWO PUMP PARALLEL FLOW INCREASE**

Modification: **N/A**

The purpose of this safety evaluation was to ensure that the performance of Reactor Water Cleanup (RWCU System Infrequent Test Procedure STP-12H, "RWCU Two Pump Parallel Flow Increase", which demonstrated the capability of the RWCU System to operate at up to 120 percent of design flow would not present an unreviewed safety question. The purpose of the test was to collect data that would demonstrate the acceptability of operating the RWCU System up to 120 percent of its current design flow. Flow would be increased in small increments while vibration, flow, motor currents, and pressures of components affected by the test were monitored.

The proposed activity had the potential to affect three accidents or operational transients evaluated in the Final Safety Analysis Report (FSAR):

- The arbitrary rupture of any single pipe up to and including complete severance of the largest pipe in the Reactor Coolant Pressure Boundary. Such rupture is assumed only if the component is subject to significant pressure.
- Reactor core coolant flow increase.
- Reactor core temperature increase.

The only Technical Specification limit potentially affected by the test was Drywell pressure. The reactor Building Closed Loop Cooling (RBCLC) System provides cooling to the drywell. Increasing the RWCU System flow put a slightly higher heat load on the RBCLC System. The Technical Specification limit would not be exceeded during the performance of the test because RBCLC System manipulations would be performed in accordance with system operating procedures which specifically address this Technical Specification limit.

This evaluation determined that the test did not increase the probability of occurrence of an accident evaluated in the FSAR. System pressures would be lowered at higher flow rates. The increase in reactor coolant flow would be negligible. The flow increase would be automatically accounted for in the reactor heat balance through the 3D Monicore System. The consequences of an accident evaluated previously in the FSAR would not be increased. The maximum differential pressure which the containment isolation valves close against would not increase. Additionally, the RWCU safety functions would be unaffected.

This safety evaluation concluded that the performance of this special RWCU System flow test did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-052 Rev. 0 AERIAL RADIATION SURVEY

Modification: N/A

The purpose of this safety evaluation was to evaluate the safety significance of an aerial radiation survey conducted by the Nuclear Regulatory Commission (NRC) at the combined James A. FitzPatrick Nuclear Power Plant and Nine Mile Point Units One and Two plants.

A risk assessment was performed in support of the safety evaluation which concluded that the safety significance to the FitzPatrick Plant from the NRC sponsored aerial radiation survey of the combined plant areas was so low as not to require consideration in the plant design basis. It further demonstrated that any probability that an accident affecting nuclear safety might result from this activity was so low as to present a negligible risk.

It was concluded that the activity presented a negligible risk and that the aerial survey did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-053 Rev. 0 SETPOINT CHANGE FOR SCREENWELL HOUSE VENTILATION SUPPLY FANS STARTING TIME DELAY RELAYS

Modification: N/A

The presently installed time relays in the Screenwell House Ventilation Supply Fans' starting circuits are Agastat 2400 Series relays and will be replaced in accordance with Modification D1-94-064. Calculation JAF-CALC-SWC-02214 Revision 0 determined the setpoint of the new relays to be 80 seconds based on the relays' design function and uncertainties. This is 25 seconds longer than the present setpoint of approximately 55 seconds.

The accident mitigating function of the supply fans, 72FN-2A and 2B, is to limit the ambient air temperature in the Emergency Service Water System pump rooms in the Screenwell House to less than 104 degrees Fahrenheit. The fans may be manually started or they will auto-start when the ambient air temperature exceeds 80 degrees Fahrenheit.

This safety evaluation reviewed the potential impact of the time delay relay setpoint change on the ventilation system's ability to maintain required temperatures in the Screenwell House. The evaluation concluded that the additional 25 seconds for fan starting time has negligible effect on the ambient air temperature rise due to the large volume of air in the Screenwell House. Also, the setpoint change has no potential for impact on the Security Plan, the Quality Assurance and Fire Protection Programs, and does not require any change to the Final Safety Analysis Report (FSAR) or Technical Specifications.

The setpoint change to the Screenwell House Ventilation Supply Fans' starting circuit does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-054 Rev. 0 FREEZE SEAL OF CARBON STEEL PIPING

Modification: N/A

Prior to this evaluation Maintenance Procedure MP-001.06 allowed the use of the freeze seal process on 300 series stainless steel piping for temporarily forming an isolation boundary for system maintenance/modification. This evaluation authorized the revision of the maintenance procedure for the application of this process to carbon steel lines.

The freeze seal process, as evaluated for carbon steel, involves the use of a refrigerant to temporarily form an ice plug in a piping system for the purpose of forming an isolation boundary to allow maintenance/modification work to be performed. The concern involving freezing carbon steel piping is that it tends to exhibit brittle characteristics at low temperatures.

Experience in commercial and industry applications and laboratory research on this process has shown that this activity can be performed successfully and safely without pipe damage and no permanent change in pipe physical properties. Procedural precautions are taken when working with carbon steel at low temperatures to prevent static and impact forces from adversely affecting the frozen pipe.

This evaluation concluded that the application of the freeze seal process to carbon steel piping in accordance with an approved procedure did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-055 Rev.0 CONSISTENT USAGE OF SEISMIC CLASSIFICATION TERMINOLOGY IN THE UFSAR

Modification: N/A

The purpose of this safety evaluation was to show that the consistent usage of seismic design classification terminologies throughout all licensing and design basis documents would reduce the existing confusions and create a uniformity in their use. Examples of existing terminologies that were utilized in various design documents to identify structures and components that are to function both during and after a safe shutdown earthquake (SSE) were:

- Seismic Category I
- Class I Seismic Category I
- Seismic Class I
- Class I Requirements, etc.

To make these terminologies uniform the only words used to identify the structures and components that remain functional both during and after an SSE shall be "Class I". The terminology used by the United States Nuclear Regulatory Commission (USNRC) for the same purpose in Regulatory Guide 1.29, "Seismic Design Classification" is "Seismic Category I". Since this Regulatory Guide is not a James A. FitzPatrick Nuclear Power Plant licensing commitment this terminology is therefore not applicable to the FitzPatrick Plant's design documents. No existing plant systems, structures or components were affected by the usage of this consistent terminology and all future designs will be in accordance with licensing commitments.

This safety evaluation concluded that the removal of inconsistencies in the seismic classification terminologies in the Final Safety Analysis Report (FSAR) did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-058 **Rev. 0** **TEMPORARY MODIFICATION 95-164, RE-ROUTE SBGT DRAIN LINE**

Modification: **N/A**

The purpose of this safety evaluation was to evaluate the acceptability of re-routing drain lines for the Standby Gas Treatment (SBGT) System. The SBGT System drain lines are routed to the Reactor Building equipment drain sump 20 TK-69B. However, due to the abnormal chemistry parameters of the drain fluids Temporary Modification 95-164 was generated to provide a method for draining the lines to a floor drain in the East Crescent which can better handle this type of effluent. The abnormal chemistry was believed to be due to ground water intrusion into the Main Stack sump. The stack sump discharges to the SBGT System drain line through the SBGT discharge line to the stack. The ground around the stack had recently been excavated and was thought to be the cause for the intrusion of the ground water. Evaluation of the groundwater problem was continuing.

This safety evaluation determined that the temporary modification was an acceptable means to re-route the SBGT drain line. This temporary modification did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report (FSAR). All requirements for secondary containment integrity were met. If the loop seals had blown out secondary containment would not have been violated due to the size of the opening. Any failure of the loop seal would not have played a role in initiating accidents previously analyzed. The modification did not create the possibility of an accident or malfunction of a different type than any previously evaluated in the FSAR. No postulated failure of the new seal could have created a conceivable new accident.

This evaluation concluded that the implementation of this temporary modification did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-059 Rev. 0 CONTINGENCY BACKUP HEATING SYSTEM FOR OUTAGE OF HOT WATER BOILER

Modification: N/A

This safety evaluation addressed the implementation of a Contingency Heating Plan to provide backup heat to the power block in the event the Hot Water Boiler becomes unavailable to provide heat. This plan will place electrical heaters in the Reactor, Turbine, Administration, Screenwell and Radwaste Buildings in the event the Hot Water Boiler fails and the reactor is shutdown. The electrical heaters will provide heat to maintain the power block above 40 degrees Fahrenheit and to keep the Station Battery Rooms and Low Pressure Coolant Injection (LPCI) Battery Rooms above 65 degrees Fahrenheit. The heaters in the Reactor Building will be powered by in house power provided by maintenance power disconnects. Electrical generators will be outside the power block and power cables will run into the power block to provide electrical power to the heaters in the Turbine, Screenwell, Administration, and Radwaste Buildings.

The installation of this Contingency Heating Temporary Modification would not increase the probability of occurrence of an accident previously evaluated in the Final Safety Analysis Report (FSAR). Failure of the Hot Water Boiler or the Contingency Heating Plan would not result in occurrence of an accident described in the FSAR. Electrical power for the heaters will be supplied from non-safety-related buses or electrical generators and will not affect power supplies for safety-related equipment. It would not increase the consequences of a malfunction of equipment important to safety evaluated previously in the FSAR because performance of Station and LPCI Batteries and the remainder of plant equipment is unchanged.

This safety evaluation concluded that the installation and implementation of this Contingency Heating Plan did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-060 Rev. 0 TEMPORARY TURBINE BUILDING COMPONENT COOLING WATER SYSTEM

Modification: N/A

The purpose of this safety evaluation was to evaluate the acceptability of providing a temporary supply of cooling water to the Service Air System compressors and the Condensate System pump motors when the Turbine Building Closed Loop Cooling (TBCLC) System was out of service. This source of coolant would only be provided when the reactor was shutdown and other TBCLC loads were not required. The coolant water was supplied from the Normal Service Water System (SWS). The flow path was from the SWS connections on the inlet side of the TBCLC heat exchangers to the air compressors and Condensate pump motors through hoses. The return flow path was to the intake through a hose to the Circulating Water System defishing line.

All hoses and connections were made with material rated for 150 pounds pressure; all connections were pressure tested prior to use and an inspection by Operations personnel was completed once per shift to verify that no degradation occurred to the hoses. There were no safety-related electrical components along the route of the hose.

This evaluation determined that the temporary source of coolant water to the Service Air compressor and Condensate pump was acceptable and did not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report (FSAR). This temporary modification supports the operation of non-safety-related equipment only. A failure of the temporary cooling system would not have impacted any safety-related equipment either by possible flooding or lack of cooling water.

This safety evaluation determined that the installation of this temporary modification to supply coolant water to both the Service Air compressor and the Condensate pump motors did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-061 Rev. 0 USE OF INITIATION/ISOLATION BYPASS SWITCHES

Modification: N/A

The purpose of this safety evaluation was to determine if an unreviewed safety question existed for using the automatic actuation logic bypass switches installed per Modifications M1-92-111, M1-92-214, F1-92-166, and F1-92-173 to meet the requirements of the Emergency Operating Procedures (EOPs) to allow valve cycling for testing when the plant was in the cold condition. The switches were installed on valves in the Residual Heat Removal (RHR), Reactor Core Isolating Cooling (RCIC), Core Spray (CS), and High Pressure Coolant Injection (HPCI) Systems to allow operation of the valves during plant fires for system shutdown. These modifications also allowed for remote manual closure of containment isolation valves while automatic opening signals were present. The safety evaluations associated with the modifications did not address the use of these switches for occasions other than during plant fires. This safety evaluation addressed these other situations.

This safety evaluation reviewed the use of these bypass switches to override system interlocks during performance of EOPs and for testing when primary containment or the system/train is not required for plant operation. The use of the switch to achieve the required valve operation both improved personnel safety.

This review determined that with the constraints placed on the use of the switches per this safety evaluation their use did not increase the probability of occurrence of an accident evaluated in the Final Safety Analysis Report (FSAR) because the automatic actuation would only be bypassed when directed by the EOPs. EOPs were developed per the Emergency Procedure Guidelines which were accepted by the Nuclear Regulatory Commission (NRC) as not affecting the safety of the plant. Use of the switches did not increase the consequences of an accident evaluated previously in the FSAR because the actions taken in the EOPs have been determined to be an effective means to mitigate the consequences of an accident if the initiating event is not known.

This safety evaluation concluded that the use of the bypass switches to override system interlocks during performance of EOPs and for testing when the system/train is not required for plant conditions did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-062 **Rev. 0** **OPERATING PROCEDURE REVISION TO - ALLOW SHUTDOWN OF CONTROL ROOM & RELAY ROOM VENTILATION SYSTEMS FOR MAINTENANCE**

Modification: **N/A**

The purpose of this safety evaluation was to evaluate the condition of shutting down the Control Room or Relay Room Ventilation Systems to permit maintenance activities during plant power operations. The Final Safety Analysis Report (FSAR) states that these systems must operate at all times during normal, shutdown, and design basis accident conditions. This safety evaluation is the basis for determining that shutdown of these systems does not involve an unreviewed safety question.

There are two primary technical issues which arise when the ventilation systems are shutdown: temperature and the ability to maintain a positive pressure in the Control Room. Maximum room stabilization temperatures and the heat up rates were measured. Sufficient margin exists to predict approach to safety limits and initiation of cooling to the area prior to reaching the limit.

The Control Room will not be pressurized to the 0.125 inches of water while the ventilation system is secured as stated in NUREG 0737. References for the safety evaluation find that the Control Room can be maintained habitable without isolation following any accident provided the reactor coolant activity is less than 0.2 micro Ci/gm.

A revised Habitability Calculation for the Control Room (Calculation JAF-CALC-RAD-00028, Rev.0, 4/18/94 "Control Room Post Accident Radiological Habitability - Assessment of Current Ventilation System Configuration") assumes a failed fuel reactor coolant (I-131) activity of 0.2 micro Ci/gm. Administrative procedures direct the plant to review the activity level (and determine that it is below 0.01 micro Ci/gm) prior to entry into a ventilation system Technical Specifications Limiting Condition for Operation (LCO).

This safety evaluation concluded that the temporary shutdown of the Control Room and Relay Room ventilation Systems for short duration maintenance activities does not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-063 Rev. 0 OPERATION OF FOUR RHR PUMPS IN TORUS COOLING FLOW MODE

Modification: N/A

The purpose of this evaluation was to determine if operation of both trains (4 pumps) of the Residual Heat Removal (RHR) System for ten (10) hours in the Torus Cooling Flow Mode (to demonstrate a design capability) posed an unreviewed safety question.

This activity was performed at the James A. FitzPatrick Nuclear Power Plant in response to United States Nuclear Regulatory Commission (USNRC) Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal Pump Strainer While Operating in Suppression Pool Cooling Mode". An event at another nuclear station involving the clogging of torus suction strainers while in normal RHR Torus Cooling system operation required the FitzPatrick Plant to adequately demonstrate that the condition did not exist at this plant.

Plant engineering recommended performing an RHR Suppression Pool maximum flow run for ten hours with all four pumps in-service. This was determined to be the limiting operating scenario most conducive to strainer clogging. This conclusion was based on a review of Emergency Core Cooling System (ECCS) flow rates, evaluation of strainer design data, and industry operating experience. The operation of the system in this manner was performed on November 7, 1995 as directed by existing plant operating procedures consistent with the license and design basis.

This evaluation concluded that the RHR performance test did not increase the probability of occurrence of an accident evaluated in the Final Safety Analysis Report (FSAR) because the RHR System was operated as designed and described in the FSAR per existing plant operating procedures. The operation of the system in this manner did not increase the likelihood of accidents described in the FSAR because in the event of a design basis accident the RHR System would have performed its Low Pressure Coolant Injection and Containment Spray functions as described in the design basis. The RHR System was not operated in a manner that violated or resulted in a change to the Technical Specifications.

This safety evaluation concluded that the test evolution did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-064 **Rev. 0** **REPAIR OF PINHOLE LEAK IN RESIDUAL HEAT REMOVAL SERVICE WATER PIPING**

Modification: D1-95-083

The purpose of this evaluation was to determine the acceptability of repairing a pinhole leak in the "B" loop Residual Heat Removal Service Water (RHRSW) System line number 10-16"-WS-151-30B while the systwem is operable.

The scope of the modification included the welding of a one inch sockolet and capped nipple over the pinhole, removing the indication by drilling a one inch or less hole in the pipe and then welding a nipple and pipe cap to form the pressure boundary. Piping in the area was ultrasonically tested and the wall thickness was evaluated and found to be acceptable.

The modification piping materials were consistent with the piping specification for class 151 pipe. The repair was performed in accordance with American Society of Mechanical Engineers (ASME) code repair criteria.

The evaluation determined that the repair activity during system operation did not increase the probability of occurrence of an accident evaluated in the Final Safety Analysis Report (FSAR) because the RHRSW System would still have performed its required safety function. The repair did not increase the consequences of an accident evaluated previously in the FSAR because the effect of the addition of the one inch leak path during the repair activity in the RHRSW piping had been shown to be bounded by previous analyses that addressed the structural integrity of the piping, secondary containment, and potential flooding effects.

This safety evaluation concluded that the pinhole leak repair as outline in Modification D1-95-083 did not constitute an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-065 Rev. 0 AUXILIARY BOILER ROOM FLOOR DRAIN PLUGS

Modification: N/A

The purpose of this safety evaluation was to evaluate the acceptability of plugging ten (10) floor drains in the Auxiliary Boiler Room per Temporary Modification 95-169. Moderate leakage existed in the Auxiliary Boiler Room. The intention of plugging the drains was to prevent this potentially contaminated liquid waste from entering the floor drains and transferring the contamination to the oil water separator. With the floor drains plugged the principal means of disposal of liquid wastes generated in the Auxiliary Boiler Room, as well as fire suppression water, is unavailable. The concerns with a loss of drainage are the potential for significant buildup of standing water, migration of the water to other areas, and possible equipment damage.

The Auxiliary Boiler Room is equipped with dike walls in front of the fire doors providing containment for 17,540 gallons of liquid. The equipment drains in the Auxiliary Boiler Room, which extend 4 inches above the floor level, will not be plugged. These equipment drains will provide drainage in the event of significant buildup of liquids in the room. The equipment drains discharge to the oil separator pit which is pumped to a liner tank before being processed through Radwaste. These provisions will prevent migration of liquid waste or fire suppression water out of the Auxiliary Boiler Room prior to detection of a leak in the room or a fire suppression actuation. These provisions will also minimize the potential for equipment damage due to liquid buildup in the room.

Equipment damage due to liquid buildup in the Auxiliary Boiler Room is not a safety concern because no safety-related nor equipment important to safety is located in the room. Floor loading due to the potential addition of 17,540 gallons of liquid is not a concern because the room rests on a concrete slab at ground level.

Additional concerns regarding the plugging of the drains were raised during the Code Review associated with the new Auxiliary boiler installation. As a result of the Fuel Oil Storage Tank Conversion modification (F1-92-146) the concerns of the Code Review have been addressed by the installation of safeguards that provide sufficient protection from a large fuel oil spill in the room.

This safety evaluation concluded that the installation of plugs did not involve an unreviewed safety question pursuant to 10 CFR 50.59.

JAF-SE-95-067 Rev. 0 ZINC INJECTION SYSTEM

Modification: N/A

The purpose of this safety evaluation was to evaluate the acceptability of operating the Zinc Injection metering pumps using only demineralized water. Normally the zinc injection pumps inject a solution of demineralized water and zinc into the Feedwater piping as stated in the Final Safety Analysis Report (FSAR). This evaluation also evaluated the acceptability of isolating the Zinc Injection System when the reactor is at rated temperature and there is Feedwater flow to the vessel.

The purpose of the zinc injection program is to reduce the levels of radiation in the Reactor Coolant System through addition of small amounts of ionic zinc.

Periodically it is necessary to operate the Zinc Injection System utilizing only demineralized water. This would be necessary to determine if the pumps are operating correctly or if a problem exists with zinc oxide solution that is being pumped.

Per safety evaluation JAF-SE-88-138, "the injection system shall be in operation whenever the reactor is at rated temperature and there is Feedwater flow to the vessel." Unavoidable trips of the operating pump or the necessity to shut down the system due to inadequate pump output make this requirement unrealistic.

The FSAR Section 9.21, Zinc Injection Passivation, was reviewed relative to operating the zinc injection metering pumps using demineralized water and the acceptability of isolating the Zinc Injection System when the reactor is at rated temperature and there is Feedwater flow to the reactor.

The FSAR classifies the system as non-safety related. There are no detrimental thermal effects associated with injecting demineralized water at 70 degrees Fahrenheit into the Feedwater System. The maximum output of the metering pumps is insignificant when compared to normal Feedwater flow.

This safety evaluation concluded that the operation of the zinc injection metering pumps using demineralized water and isolating the Zinc Injection System when the reactor is at rated temperature does not involve an unreviewed safety question pursuant to 10 CFR 50.59.