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United States Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station O-P1-17
Washington, D.C. 20555

Subject: **James A. FitzPatrick Nuclear Power Plant**
Docket No. 50-333
Summary of Plant Changes, Tests, and Experiments for 1999/ 2000
as Required by 10 CFR 50.59

Attachment 1: Summary of Changes, Tests, and Experiments for 1999/2000

Dear Sir:

This letter transmits a summary of the changes, tests, and experiments implemented at the James A. FitzPatrick Nuclear Power Plant during 1999/2000 as required by 10 CFR 50.59(b)(2).

Provided with each summary is the Nuclear Safety Evaluation number (e.g. JAF-SE-95-0001), revision number, title, engineering change number, if applicable, followed by a brief description of the corresponding change, test, or experiment.

If you have any questions concerning this report, please contact Mr. A. Zaremba of my staff at (315) 349-6005.

Very truly yours,

(Acting)

T. A. Sullivan

TAS:KAM:las

cc: USNRC Regional Administrator
USNRC Office of the Resident Inspector
USNRC Project Manager

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ATTACHMENT I

Summary of Changes, Tests, and Experiments for 1999/2000

Introduction to the 1999/2000 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 it's 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-89-042, REV. 1: **Addition of Oil Sampling Capability to RHR, RHRSW, and Core Spray Pump Motors**

ENGINEERING CHANGE: **M1-89-086**

Revision 1 of JAF-SE-89-042 evaluates elimination of the requirement to lock shut RHR, RHRSW and Core Spray pump motor bearing lube oil reservoir sampling valves which was established by modification M1-89-086. Use of locking devices to control valve position was a conservative restriction not required by industry standards or regulations. Eliminating the requirement to lock valves in position is in accordance with ALARA and FME practices, while not significantly changing the risk associated with valve mispositioning. To drain an oil reservoir, an end cap must be removed and a drain valve opened, so multiple errors must occur to inadvertently empty the pump of oil.

JAF-SE-90-074, REV. 3: **Temporary Chlorine Water Treatment System for the JAF Service Water Systems**

ENGINEERING CHANGE: **F1-90-038**

This modification to the Water Treatment System will provide a temporary chlorine water treatment system for the Normal Service Water, Emergency Service Water, RHR Service Water and Fire Protection systems to mitigate fouling due to zebra mussels.

No unreviewed safety question exists as defined in 10 CFR 50.59.

JAF-SE-91-051, REV. 1: Thermal Overload Configuration for Motor Operated Valves

ENGINEERING CHANGE: M1-90-174

The purpose of the proposed activity is to provide thermal overload protection for the MOVs as listed in the modification package M1-90-174. The proposed activity (Modification M1-90-174, Rev.1) will replace the existing 300% thermal overloads with the thermal overloads sized in accordance with the IEEE -741 guidelines.

No unreviewed safety question exists as defined in 10 CFR 50.59.

JAF-SE-91-061, REV. 6: Replacement of Control Switches for 10MOV-12-A/B

ENGINEERING CHANGE: M1-91-156

JAF modification M1-91-056 replaced the existing keylock open/closed Control Room switch on the 09-3 panel with a three position open/stop/closed switch for the RHR Heat Exchanger outlet valve (10MOV-12A/B). This allows remote throttling of 10MOV-12A/B from the Control Room

This safety evaluation documents the evaluation of the effect of this change on the eleven modes of operation of the RHR system including the continued, limited use of valve 10MOV-27A/B as a throttle valve in the post accident LPCI mode of operation. There is no unreviewed safety question as defined in 10 CFR 50.59.

JAF-SE-94-052, REV. 2: Support MSIV PNS Testing

ENGINEERING CHANGE: N/A

The proposed temporary modification installs 2 jumpers, one around the Steam line B MSIV 90% position switch contacts in the "A1" RPS Scram sensor trip channel and one around the steam line B MSIV 90% position switch contacts in the "B2" RPS Scram sensor trip channel. These jumpers will be in series with the supply power fuses and the trip channel relay. When the fuses are installed per TST-31, (a revised temporary test procedure for ST-1I), the appropriate sensor trip channel will be made inoperable due to it being untrippable. TST-31 provides direction on the appropriate Technical Specification actions required to address the inoperable sensor channel.

This temporary modification is required to support MSIV Closure Scram functional testing. In our current plant condition (RUN mode) with a failed inboard MSIV 90% position switch 29PNS-80B, a full scram would result if MSIV 80A or MSIV 86D were stroked to the 90% open position surveillance during testing performed by ST-1I. This temporary modification will prevent a trip signal from the failed position switch sensor trip logic when its power supply fuse is installed per TST-31 to allow for testing of MSIV 80A and MSIV 86D without producing a full scram.

TST-31 has been revised to provide strict guidance for test performance and actions to be followed. With one MSIV position sensor disabled and made incapable of tripping by the use of these jumpers and TST-31, the MSIV Closure Scram function is maintained by the remaining operable sensor channels. Therefore, an unreviewed safety question does not exist.

JAF-SE-94-072, REV. 3: EDG Fuel Oil Storage Tank Leak Tightness Testing STP-93P

ENGINEERING CHANGE: N/A

The proposed activity will test 93TK-6A,B, C, and D for required leak tightness per DEC Regulations. The performance of the test will require an underground fuel oil storage tank be isolated for testing, rendering that tank inoperable for the duration of the testing. A planned 7 Day LCO will be entered in accordance with AP-05.13 and Technical Specification 3.9.B.3 due to the unavailability of a sufficient quantity of fuel oil. The testing will be performed on one tank at a time, therefore one of the two redundant trains will always be operable at all times during testing. For the out of service train both diesels have the option of cross-tying to the available underground fuel oil tank. If no maintenance is performed during the scheduled LCO this will allow for both diesels to be considered functionally available. One underground fuel oil tank will provide enough fuel for 3.5 days of loaded operation for a train of Emergency Diesel Generators. If maintenance is performed on one of the two diesels then the cross-tying maybe used to keep the other diesel available. There is no unreviewed safety question per 10 CFR 50.59.

JAF-SE-94-114, REV. 2: 46MOV-102B In-Situ Design Bases DP Test

ENGINEERING CHANGE: N/A

This Nuclear Safety Evaluation analyzes the impact of performance of SIP-46I and STP-46J and ST-8XD on plant safety. These Special Tests are being performed to demonstrate the operability of 46MOV-101A(B) and 46MOV-102B under conditions of design basis differential pressure, as required by NRC Generic Letter 89-10. Valve operator characteristics will be monitored and recorded during test performance. The ESW system will be operated within its design basis limits and Technical Specification requirements regarding ESW system and EDG system operability will be adhered to during test performance. Based on the above information, it is concluded that performance of these Special Tests does not constitute an unreviewed safety question as defined by 10 CFR 50.59.

JAF-SE-95-055, REV. 2: Consistent Usage of Design Classification Terminology in the UFSAR

ENGINEERING CHANGE: N/A

Revision 0 of this Nuclear Safety Evaluation attempted to clarify seismic design classification terminology in the UFSAR. The proposed change was to make consistent usage of one terminology for "Seismic Classification". This was to be accomplished by revising the UFSAR to indicate "Class I" wherever it stated the following: "Seismic Category I", "Class I Seismic Category I", "Seismic Class I", and "Class I Requirement". The NSE concluded that the terms were for all practical purposes synonymous since all "Class I" systems, structures, and components were analyzed and designed to function both during and after a Design Basis Accident (DBA), and as such were designed for "Seismic" loading.

In summary, the removal of existing inconsistencies in the design classification terminology for structures and equipment in the UFSAR is acceptable. The use of consistent design classification terminology as described in this evaluation does not adversely affect any structure, system or component (SSC) assumed to operate during normal plant operation or an accident. No unreviewed safety question exists as defined by 10 CFR 50.59.

JAF-SE-96-013, REV. 3: Refuel Bridge Upgrades

ENGINEERING CHANGE: M1-95-122

The modification includes a Programmable Logic Control (PLC) which will control operations of the refuel bridge, trolley and main hoist. The modification includes changes to the controls governing main hoist operation. Procedural controls for moving irradiated fuel using the "fuel grapple" device only will not be affected by this modification.

This safety evaluation addresses modifications to the Refueling Bridge Platform to upgrade its ability to safely support refueling floor operations. It shows that the Refueling Bridge Upgrades (minor modification No. M1-95-122) do not constitute an unreviewed safety question.

JAF-SE-96-016, REV. 4: **Acceptability of Establishing a One Hour Post Accident Operating Time Requirement for the ADS System to Mitigate LOCA**

ENGINEERING CHANGE: **N/A**

Revision 4 of JAF-SE-96-016 clarifies the basis of acceptability of a one hour post-accident operating time for SRV SOV's after loss-of-coolant accidents with break areas $\geq$ 0.5 ft^2. The JAF SAFER/GESTR LOCA analysis was reviewed to establish that the RPV would be depressurized to less than 50 psig (the minimum opening pressure for SRVs when remotely operated) within one hour. It was also demonstrated that liquid flow through a 0.5 ft^2 break is capable of removing core decay heat one hour following an accident. Therefore, SRVs are not required to maintain the RPV depressurized (to allow low-pressure ECCS to feed the RPV) for a duration longer than one hour following a LOCA with break area $\geq$ 0.5 ft^2.

JAF-SE-96-045, REV. 3: Power Uprate Startup Test Plan

ENGINEERING CHANGE: M1-96-061

The power uprate startup test plan was implemented during startup from refueling outage 12. The LEFM test plan shall be completed during cycle 14. The scope of testing to be performed is documented in an approved engineering report and has been found acceptable. The individual procedures to be used to implement the test plan have been reviewed and found acceptable and are discussed in the engineering report (NYPA report no. JAF-RPT-MULTI-02420 "Power Uprate Startup Test Program Plan, revision 6). As noted in 5.2 above POT-06B will be approved and issued upon completion of this NSE. The test requirements specified in Supplement 1 to reference 7 ensure system performance at a higher actual power (as a result of using the LEFM to improve accuracy in the feedwater flow measurement) is acceptable. Since the plant is operated within analyzed limits during performance of this testing, and no previously unanalyzed test methods are introduced, this testing may be performed in accordance with 10 CFR 50.59.

JAF-SE-96-071, REV. 2: **Impact of Increased Isolation Time of Reactor Building Ventilation System on FSAR Analyzed Events**

ENGINEERING CHANGE: **N/A**

The purpose of this safety evaluation is to determine whether any unreviewed safety question exists for the FSAR and TS changes submitted.

Revision 2 of this NSE incorporates the results from the latest radiological calculations for the Fuel Handling Accident. It specifically addresses the isolation requirements of the Reactor Building and the Control Room as applicable from the latest analyses. This activity was performed to support outage planning to ensure that the Emergency Service Water (ESW) unavailability during the outage will not adversely impact the Control Room doses.

The proposed changes to the FSAR provide:

1) An accurate description of the Reactor Building Isolation and Control System functions, and
2) For a small fraction (8 second release plus minor exfiltration) of the Fuel Handling Accident source term to be released unfiltered at ground level before the Reactor Building isolates and for the majority of the source term to be released (over 2 hours) filtered by the Standby Gas Treatment System and out the Main Stack as an elevated release.

The proposed change to Technical Specification Bases 3.8 is editorial. It does not change the bases of the technical specification in that most of the radioactivity from a fuel handling accident is released filtered and elevated. The change accurately describes the location of the radiation monitors as external to the refuel floor ventilation ductwork.

Although a small fraction of the source term (8 second release plus minor exfiltration) from a Fuel Handling Accident is released through the refuel floor ventilation system prior to reactor building isolation, the majority of the activity is released (over 2 hours) via the Standby Gas Treatment System and out the Main Stack as a filtered, elevated release.

The release of this activity through the refuel floor ventilation exhaust is in accordance with RETS Bases 3.8 because most of the activity is released to the environment via the Standby Gas Treatment System and the Main Stack. Even though a fraction of the activity is released via the reactor building refuel floor exhaust system, the dose consequences from a Fuel Handling Accident are still below 25% of the 10CFR100 limits and do not cause an overexposure to control room personnel.

Administrative controls are to be used to pre-isolate the Control Room if the conditions for a Fuel Handling Accident could cause overexposure to control room personnel and for ALARA purposes.

The dose to receptors in the low population zone and the site boundary are less than 25% of the 10CFR100 dose guidelines. The dose to receptors in the control room does not cause an overexposure. The criteria for the Standard Review Plan, 10CFR100, and General Design Criteria 19 are met. Therefore, an Unreviewed Safety Question does not exist.

JAF-SE-97-005, REV. 2: Feedwater Flow Ultrasonic Monitoring System (LEFM)

ENGINEERING CHANGE: M1-96-061

The LEFM system will be used to establish a correction factor (CF) to the existing feedwater flow value input into 3D Monicore to compensate for the fouling bias at the feedwater venturis. This will allow the plant to operate closer to the licensed core thermal power limit. Feedwater flow indication input into 3D Monicore is higher than actual due to the fouling bias error. Compensating the feedwater flow input into 3D Monicore will be a manual process. The CF will be manually entered into EPIC which corrects the feedwater flow value input into 3D Monicore. Determination of the CF will be controlled by RAP 7.3.38 and will consist of a process to validate LEFM flow values. Trending and analysis of LEFM data will be performed to ensure LEFM system health.

Installation of the LEFM system has no affect on any plant component or system. The feedwater flow CF values will not affect the feedwater control system. Installation will have no affect on any equipment important to safety identified in the SAR nor will it affect any accident analysis.

JAF-SE-97-021, REV. 3: Installation of Maximum Density Spent Fuel Racks in the Pool

ENGINEERING CHANGE: F1-96-008

This modification includes the installation of three (3) max-density racks in the SFP. The cell locations in these racks, in addition to the utilization of the peripheral locations of the existing racks that were inaccessible with the refuel bridge (106/113 locations) will provide enough number of locations to accommodate the discharge from R013 plus full-core off-load capacity.

If the dry cask storage facility (ISFSI) is not in place by 2001, four smaller racks designated as F1 through F4 will be fabricated and installed to accommodate the spent fuels providing for continued operation of JAF until 2004. Note, that the full-core off-load capability will be maintained.

The installation of one or more of the seven racks in accordance with the plant schedule, between 1998 and 2002, does not possess any concerns for the safe operation of the plant and does not pose any threat to the health and safety of the general public.

JAF-SE-97-023, REV. 0: Elimination of Fire Detection Capability on the Refueling Floor

ENGINEERING CHANGE: N/A

This NSE demonstrates that eliminating the existing smoke detector capability does not decrease fire safety because administrative limits on allowable combustibles will ensure that potential fire severity does not impact Reactor Building structural integrity. In addition, it has been demonstrated that a Refueling Floor fire will not adversely affect the ability to achieve and maintain safe shutdown or to provide fuel pool cooling either at or above cold shutdown.

FSAR Section 9.8.3.3 will be revised to clarify references to fire detection capability on the Refueling Floor of the Reactor Building.

Approved fire protection program documentation will be revised to reflect the change in the plant configuration for abandoning in place the smoke detectors on the Refueling Floor. There is no increase in the consequences of any fire or any other event due to the elimination of the smoke detectors on the Refueling Floor. The elimination of this smoke detection capability will not initiate any event nor directly affect any equipment important to safety. Safe shutdown capability of the plant will be maintained. Structural integrity of the Reactor Building, Reactor Building Crane, and the Refueling Platform will be ensured. Furthermore, it has been demonstrated that vessel and fuel pool cooling capability during shutdown and refueling outages will be available in the unlikely event of a Refueling Floor fire. Therefore, NRC acceptance criteria for the fire protection program has been maintained and plant safety will be improved as a result of program enhancements to control combustibles and to address off-normal vessel and fuel pool cooling requirements.

JAF-SE-97-036, REV. 1: **Filtering of Emergency Diesel Generator Fuel Oil Storage Tanks 6A & 6D**

ENGINEERING CHANGE: **N/A**

Filtering the contents of the EDG underground fuel oil storage tank will be performed using a vendor supplied filtration skid. Performing this activity during plant operation does not render the EDG inoperable. Adequate inventory will be maintained in the underground tank throughout the filtering activity. In the event of vendor equipment failure or severe weather (heavy rain, lightning, high winds, etc) the filtering activity will be terminated until conditions permit restart of the activity. The filtering equipment will be staged in the diked area designed to accept a fuel oil spill, to minimize the consequences of a spill. In the event of an EDG start demand, filtering activities will be halted and the system restored to normal.

JAF-SE-98-006, REV. 1: Cycle 14 Reload Core

ENGINEERING CHANGE: N/A

This evaluation documents the changes made to the reactor core by the refueling prior to the start of Cycle 14. The safety analyses performed on the Cycle 14 core have been reviewed and found adequate in scope and content against the requirements in Reference 1 and the FitzPatrick SAR and Technical Specifications. Fuel design changes have been reviewed against designs previously used at FitzPatrick. On the basis of this review no unreviewed safety question exists, and the safety analyses properly support the limits established in the Core Operating Limits Report (COLR).

Based on the foregoing summary it can be concluded that operation of FitzPatrick in the Cycle 14 core as given in the reference documents, to the limits given in the COLR does not constitute an unreviewed safety question.

The fourteen replacement control rod blades(CRB) installed under this modification have been evaluated against the SAR. There are no Technical Specification requirements affected by this modification. Design differences between these CRBs, other replacement CRBs and the original equipment were found to be insignificant to their interface with other plant components and their function. Based on these conclusions the replacement CRBs are acceptable.

JAF-SE-98-013, REV. 3: **Residual Heat Removal and Core Spray Suppression Pool Suction Strainer Replacement**

ENGINEERING CHANGE: **F1-97-031**

Modification F1-97-031, Rev. 0, "Residual Heat Removal and Core Spray Suppression Pool Suction Strainer Replacement", installed new suppression pool suction strainers for Residual Heat Removal and Core Spray system pumps. The replacement-strainers are substantially larger than the original strainers and provide additional margin for debris loading following a design basis accident. Consistent with the original licensing basis for FitzPatrick, the replacement strainers have been evaluated to ensure adequate NPSH would be available to the pumps, under the most limiting conditions analyzed in the current LOCA analysis, but with the worst case debris loading determined in accordance with the criteria of Reg. Guide 1.82, Revision 2. The strainer assemblies and supports have been analyzed and qualified for seismic and Mark I hydrodynamic loads specified in the FSAR and Plant Unique Analysis Report.

A re-evaluation of the long-term suppression pool temperature following a design basis Loss-of-Coolant- Accident, developed by General Electric for FitzPatrick, demonstrated additional ECCS pump NPSH margin. The new containment analysis in GE-NE-T23-00766-00-01 decreased the analytical peak suppression pool temperature to 196.3°F.

A temporary construction opening was made in the torus to facilitate personnel and equipment movement for the strainer installation. Integrity of the repaired opening was ensured by qualification and non-destructive examination of the weld, and by post modification testing. A primary containment pressurization test was performed at peak accident pressure to verify leak-tightness at the weld and structural integrity of the repaired area.

JAF-SE-98-014, REV. 0: **Revision to OP-54 to Operate Radwaste Building / Tank Exhaust Fans with Building Supply Secured for Maintenance**

ENGINEERING CHANGE: **N/A**

The purpose of this NSE is to address and evaluate a revision to OP-54, Radwaste Building Heating and Ventilating System to allow for the Radwaste Building Supply fans to be secured for maintenance while providing exhaust airflow. FSAR section 9.9.3.5 describes normal system operation as having one building supply fan in service with a redundant supply fan in standby.

Securing the Building Supply Fans for maintenance while requiring the Building Exhaust Fans to be secured with one Tank and Equipment Fan in service provides 3600 cfm of exhaust flow. The Radwaste Building pressure requirement is to maintain a slight negative pressure to atmosphere. System operation of a Tank & Equipment Exhaust Fan with no makeup air provided by the building supply fans allows for negative building pressure with respect to atmosphere to be maintained. This NSE does NOT affect system operation as described in the FSAR of a filtered, monitored release to the environment and system response will remain to automatically shutdown the operating Tank and Equipment exhaust fan upon detection of high levels of radiation in the exhaust plenum. As demonstrated in Review and Analysis section, adequate margin exists to maintain the required space temperatures.

The Radwaste Building HVAC System is not required to operate under any Design Basis Accident nor does it have any safety-related functions. Based upon the above evaluation, an unreviewed safety question does not exist.

JAF-SE-98-015, REV. 1: **Installation of HPCI / RCIC Package Boiler and Associated Piping**

ENGINEERING CHANGE: **F1-97-034**

Modification F1-97-034 involves the use of a temporary boiler and the installation of piping, pipe supports and valves to establish a steam supply flow path for HPCI and RCIC turbine slow roll testing and overspeed testing. The new piping associated with this modification goes through the Nitrogen Storage Building and into the Reactor Building through penetration S-733. It ties in to the existing HPCI and RCIC steam supply lines downstream of the outboard containment isolation valves, and uses a spectacle blind flange/spacer ring arrangement to establish and isolate the flow path. The piping from the steam supply tie-in to the upstream flange adjacent to the spectacle flange is designed to the same classification as the respective HPCI and RCIC steam supply lines. All installed piping is seismically analyzed. The temporary boiler has been specified to provide the appropriate steam parameters including steam quality to perform the testing (Ref. 7.3). This modification does not involve an unreviewed safety question, nor does it require a change to JAF Technical Specifications.

JAF-SE-98-025, REV. 2: **High Pressure Coolant Injection and Reactor Core Isolation Cooling Suppression Pool Suction Strainer Replacement**

ENGINEERING CHANGE: **F1-98-100**

Modification F1-98-100, Rev. 0, "High Pressure Core Injection and Reactor Core Isolation Cooling Suppression Pool Suction Strainer Replacement", installed new suppression pool suction strainers for HPCI and RCIC system pumps. The replacement strainers are larger than the original strainers and provide additional margin for debris loading following a small or intermediate break loss-of-coolant-accident. The strainer assemblies have been analyzed and qualified for seismic and Mark I hydrodynamic loads specified in the FSAR and Plant Unique Analysis Report.

A review of the modification in accordance with 10CFR 50.59 concluded that the modification does not increase the probability or consequences of an accident or of a malfunction of equipment important to safety previously evaluated in the safety analysis report. Further, the possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report has not been created. The margin of safety as defined in the Technical Specification Bases is not reduced and no Technical Specification change is required. This change, therefore, does not involve an unreviewed safety question.

JAF-SE-98-029, REV. 1: **Noble Metals Chemical Addition/NMCA/Noble Chem™ Application Procedure for JAFNPP**

ENGINEERING CHANGE: **M1-97-111**

Modification M1-97-111 addresses the deposition of noble metals to the wetted surfaces within the reactor pressure vessel. The NobleChem™ process developed by GENE (generally referred to as Noble Metals Chemical Addition (NMCA)) will be controlled using Temporary Operating Procedure, TOP-300, "Noble Metals Chemical Addition" and GENE application procedure GE-NE-P86-0002-04-01, "NobleChem™ Process Application Procedure for JAFNPP."

NobleChem™ treatment of reactor internals and piping at JAF and lowering the hydrogen injection rate will reduce the potential for IGSCC and decrease the radiological dose associated with generation of ^{16}N. Consequently, due to its mitigation of the effects of IGSCC and requirement of lower hydrogen injection, NobleChem™ treatment will increase the lifetime of the JAF reactor internals and piping and reduce radiation exposure to JAF personnel, both of which are beneficial to plant operation and performance.

NobleChem™ treatment does not involve an unreviewed safety question nor does it require a change to JAF Technical Specifications.

JAF-SE-98-039, REV. 0: **Reactor Building Closed Loop Cooling Water Pump Motor Speed**

ENGINEERING CHANGE: **N/A**

The existing description in the Final Safety Analysis Report (FSAR) Table 9.5-2 identifies the RBCLCW Pump motor speed as 1750 RPM. This evaluation involves changing the FSAR speed for the RBCLCW pump motor from 1750 RPM to 1800 RPM (1750 rpm minimum). This change to the FSAR does not involve:

a) A change to the RBCLCW System interfaces, b) A negative impact to system performance, or c) A change to previous evaluations at JAF.

The resultant increase in motor speed, as described within this safety evaluation, will not result in rapid Changes in the RBCLCW water temperature or flow to the drywell unit coolers. Therefore, there will be no drywell pressure transients resulting from:

- Raising RBCLCW temperature or lowering drywell cooler RBCLCW flow, which could raise drywell pressure, resulting in a high drywell, pressure alarm or high drywell pressure reactor scram.
- Lowering RBCLCW temperature or raising drywell cooler RBCLCW flow could lower drywell pressure, resulting in a drywell to Torus differential pressure going below the Technical Specification LCO limit of 1.7 psid.

This change is merely an equipment description correction for original plant equipment and poses no physical change to the facility. As a result of this evaluation there exist no Unreviewed Safety Questions (USQ).

JAF-SE-98-040, REV. 1: De-Energizing Drywell Cooling MOVs (TM 98-063)

ENGINEERING CHANGE: N/A

The existing circuits for non-safety related components 15MOV-100A1 / A2 / A3 / A4 / B1 / B2 / B3 / B4 has been identified as being susceptible to a single failure that may potentially effect QA Category 1 safety related containment penetrations. A postulated short circuit due to a failure of one of these non-safety related valves, with the single failure of the circuit breaker in the same circuit, could overheat the cable and possibly cause damage to the electrical penetration.(Re: DER-98-01910).

Since the station is currently susceptible to this potential scenario, the eight MOVs will be de-energized, until a modification can be installed or the configuration has been determined to be acceptable. The purpose of this evaluation is to show that temporarily opening breakers that supply power to these valves does not present the station with an unreviewed safety question.

The MOVs and their associated indicating switches are used to provide cooling water flow and flow isolation to the individual cooling coils on the non-safety-related drywell coolers. Temporary de-energization of these components does not adversely affect the operation of any equipment important to safety. Furthermore, sufficient redundant indications exist such that the MOVs and their indicating lights can be de-energized without compromising the safe operation of the plant.

JAF-SE-98-044, REV. 1: **Permanent Installation of a Valve for Outage Condensate Demineralizer Water Moves**

ENGINEERING CHANGE: **JD-97-087**

A manual valve will replace a spectacle flange near the location that had a remote manually operated AOV in the original plant design. This accessible valve will be used to support outage water moves.

The proposed activity more closely represents the original design condition and will not increase probability or consequences of adverse situations.

The proposed activity has no effect on the various other programs currently being evaluated under this process. None of the above actions results in an unreviewed safety question.

JAF-SE-98-051, REV. 1: **Inboard Main Steam Isolation Valves (MSIV's) NAMCO Connectors**

ENGINEERING CHANGE: **N/A**

The proposed activity being evaluated consists of accepting the installation as is for the plug-in cable supplied with Namco EC210 Shielded Cable Seal/Connector Assemblies for the MSIV SVCA, located inside the primary containment. Approximately less than 3 feet of the plug-in cables are not in conduit. UFSAR Section 7.1.9.a states that all cables inside containment are installed in conduit.

An unreviewed safety question is not involved because the plug-in cables installed in free air are environmentally qualified and the use of conduit for the subject installation is not required to assure the functionality of the MSIVs. Additionally, no requirements exist in the JAF Electrical Separation Criteria, 10CFR50 Appendix R or in the Accident Analysis that requires the use of metallic conduit for the subject installation.

JAF-SE-99-001, REV. 1: Alternate Hydrogen Supply

ENGINEERING CHANGE: N/A

The existing hydrogen supply has been rendered inoperable due to a fire. In order to restore the Hydrogen Addition System, an alternate supply of hydrogen will be provided to the Station. A pressure control cabinet will be mounted on the hydrogen pad located on the West side of the site. Piping will be routed from the control cabinet to the Station hydrogen supply piping. A trailer containing tanks of hydrogen will be connected to the control cabinet. Also, a connection will be made to the reserve bank located on the West end of the hydrogen pad. This will provide additional capacity beyond the tanks on the trailer. The reserve bank was not damaged during the fire. Upon installing and testing this equipment, the system will be lined up such that hydrogen will be supplied to the Station.

Installation and testing of the hydrogen supply piping will be accomplished using the guidance of EPRI Report No. NP-5283-SR-A, Guidelines for Permanent BWR Hydrogen Water Chemistry Installations - 1987 Revision. The pressure control station with its two full-flow pressure reducing regulators will maintain the required hydrogen pressure entering the plant. At least one excess flow check valve with a stop-flow setpoint no greater than 100 SCFM will restrict flow in the event of a line break. The hydrogen storage facility is located approximately 250 feet from the nearest safety-related structure (Diesel Generator Building). This distance will provide sufficient protection for safety related structures in the event of a hydrogen explosion.

The piping and equipment inside the plant will remain the same. The hydrogen injection rate into the Condensate System will not be changed. Hydrogen injection rates will continue to be less than or equal to 19 SCFM. There will be no increase in the probability or the consequences of an accident or a malfunction of safety related equipment. This activity will not create any new accidents or failure modes. The margin of safety as defined in the basis' for the Technical Specifications will not be reduced.

JAF-SE-99-002, REV. 2: Reactor Building Crane Upgrade

ENGINEERING CHANGE: F1-91-270

This modification upgrades the Reactor Building Crane to meet the criteria for a single-failure- proof crane as outlined in NUREG-0554. The upgraded crane components and controls is described in the Ederer Topical Report EDR-1(P)-A. The Ederer Topical Report has been accepted by the NRC to comply with the requirements of NUREG-0554. The upgrade will produce a crane with a safer operation and the ability to handle critical heavy loads without the possibility of a single failure of a component causing the loss of load control. This evaluation demonstrates that this modification to the Reactor Building Crane does not constitute any unreviewed safety question as defined by 10 CFR 50.59 nor does it involve significant hazards defined in 10 CFR 50.92. Therefore, the activity can be performed as proposed.

JAF-SE-99-003, REV. 0: **Use of LPCI Initiation / Isolation Bypass Switches by Procedures to Support EOP Actions**

ENGINEERING CHANGE: **N/A**

Bypass switches affecting LPCI valves (installed per modification M1-92-111) and their associated valves are listed below. The switches are a two-position keylock design, the positions being Norm and Bypass. With the switches in normal, the valves operate as they did previous to these modifications. With the switches in bypass, all automatic opening and closing functions (including the low pressure permissive) are bypassed and remote manual operation from the normal control switch in the control room is allowed.

SWITCH	VALVE(S)
10A-S22A	10MOV-25A
10A-S22B	10MOV-25B
10A-S23A	10MOV-27A
10A-S23B	10MOV-27B

The switches will be used in support of the EOPs to carry out desired actions such as terminate and prevent injection or to permit LPCI injection to the RPV to mitigate the consequences of the situation at hand. The use of the switches in these situations (terminate and prevent injection and override) is beyond the design bases, thus bypassing automatic operation is acceptable. In these cases, the NRC has accepted that operator intervention per EOP guidance does not adversely affect the safety of the plant.

The actions to be taken in the EOPs and EP support procedures are generally in accordance with revision 1 of the BWROG Emergency Procedure Guidelines/Severe Accident Guidelines (EPG/SAG) which were previously evaluated in JAF-SE-98-053, Rev. 0. JAF IPE insights consistent with the EPG/SAG strategies have been implemented in EOPs and support procedures. JAFs baseline IPE was reviewed and accepted by the NRC in another SER (TAC No. M74411).

JAF-SE-99-004, REV. 1: Service Air Outage

ENGINEERING CHANGE: N/A

Instrument and breathing air supply will not be affected. Service Air is not a QA category I system and is not credited for accident mitigation. Loss of service air does not impact the plants ability to safely shutdown.

TOP-293 will temporarily remove the service air system by closing 39FCV-110 and depressurizing the system by opening service air valves in order to perform periodic maintenance. Instrument air and breathing air supply will not be interrupted. The Service Air System will be depressurized while the work is being performed which is estimated to take two shifts but may take up to two days.

Service air is not involved in any accident initiation scenario and would not increase the consequences of an accident. The Service Air System does not perform any safety-related functions and can not cause a malfunction of any safety-related equipment. The Service Air System does not change any acceptance limits listed in the Technical Specifications; therefore, it does not reduce the margin of safety. None of the above actions results in an unreviewed safety question.

JAF-SE-99-005, REV. 0: De-Energize 71RBACD5

ENGINEERING CHANGE: N/A

120VAC distribution panel 71RBACD5, located in the Reactor Building, will be de-energized to permit maintenance on its feeder breaker. This panel powers a number of non-safety related indication, control, and annunciation circuits, some of which are described in the UFSAR. This circuits include a computer used for continuous monitoring of plant chemistry parameters, the normal power supply for the Post-Accident Sampling System (PASS), annunciator power for the Fuel Pool Cooling and Cleanup System (FPCC) and for portions of the Reactor Building Ventilation System, the East Crescent area ambient temperature indicator, an Area Radiation Monitor (ARM) in the East Hydraulic Control Unit (HCU) area, and door status lights for the main access airlock through the Secondary Containment.

Various compensatory actions will be used to permit de-energizing these circuits. The chemistry computer and the PASS power supply will be made inoperable in accordance with procedures for planned LCO entries. The annunciator systems will be compensated for by frequent monitoring of appropriate parameters in the respective systems. Temporary indicators will be substituted for the disabled temperature indication and the ARM. Additional administrative controls will be instituted to ensure that at least one airlock door remains closed at all times to provide Secondary Containment Integrity.

With these compensatory actions in place, each affected system will function within the limits expected in the UFSAR in the prevention or mitigation of accidents, transients, and equipment malfunctions.

JAF-SE-99-006, REV. 1: Liquid Abrasive Decontamination Unit (LADS) Replacement

ENGINEERING CHANGE: M1-98-191

Minor Modification M1-98-191, "Liquid Abrasive Decontamination System (LADS) Replacement" will add a dry-pressure blast decontamination unit (20DCU01) as a replacement for the existing Liquid Abrasive Decontamination System (LADS) Unit in the Radwaste Treatment System. While the Radwaste System is classified as a nuclear safety-related system (the Safety Related functions of the Radwaste System and associated components are to help maintain integrity of primary containment boundary and to provide containment isolation valve position indication), the decontamination unit replacement has no impact on any safety significant function.

JAF-SE-99-007, REV. 0: **Storing Control Rod Blade and Hanger Assemblies on Spent Fuel Pool Walls**

ENGINEERING CHANGE: **N/A**

In summary, transfer of up to 30 Control Rod Blades (attached to "short" hanger assemblies) from the Control Blade Storage Rack in the SFP to a location on the wall of the SFP where a bracket attaches the Control Rod Blade and hanger assembly to the curb above the SFP wall will have no impact on the safety related functions of the SFP, the high density fuel racks, and the FPM. The evaluation demonstrates that the activity does not constitute any unreviewed safety question as defined by 10 CFR 50.59. Therefore, the activity can be performed as proposed.

JAF-SE-99-008, REV. 0: **Update of FSAR to Reflect 1973 Replacement of LPCI Pump Low Flow Switches – Range Change**

ENGINEERING CHANGE: **N/A**

The proposed change to FSAR Table 7.44 (Sheet 2 of 3) to change the range of the LPCI pump low flow switches from 0 to 800 gpm to 0 to 3650 gpm corrects a minor error that has existed since plant construction and initial commercial operation. This change will not adversely affect plant safety because this is a documentation only change and the DPIS switches are more reliable devices than the original "paddle type" flow switches. Section 7.4.3.5.4 of the FSAR correctly describes how the installed switches function. The change in range updates the FSAR to agree with the range of installed DPIS switches (original construction design change).

JAF-SE-99-009, REV. 0: Isolation of Air Supply to Automatic Purge for Off-Gas

ENGINEERING CHANGE: N/A

This activity is the removal of the automatic purge feature of the Off Gas system by Service air, accomplished by the tagging shut of the air supply valve 39SAS-88. This is necessary to eliminate a combustion source through the degraded Air Operated Valve 38AOV-115, in the event of combustion in the SJAE after condenser or Off Gas system piping.

The automatic purge feature is described in Chapter 11 of the Safety Analysis Report and is intended to facilitate maintenance on the system after system shutdown. Air is used to purge hydrogen to prevent a potentially combustible mixture inside the piping. This feature. As described in AOP-4, requires Operator action to verify that the air purge can be initiated. The actions described are a physical walk-down of the piping and components in the condenser bay. Contained in this area is the manual air supply valve 39SAS-88, therefore requiring operators to open this valve prior to purging can easily be accomplished. This is not taking credit for operator action in lieu of an automatic function since resetting the system after explosion also requires operator action.

During a normal plant shutdown once 38AOV- 113A & B close, conditions should allow the automatic purge system to initiate. This feature is not described in the SAR and has no nuclear safety function. The PTR in place notifies Operators of the need to open 39SAS-88 in the event of plant shutdown.

Based on the evaluation provided there is no unreviewed safety question as a result of this Protective Tagging Record.

JAF-SE-99-010, REV. 0: Organizational Changes

ENGINEERING CHANGE: N/A

The organizational changes proposed by this Nuclear Safety Evaluation involve the creation of the positions of Plant Manager (PM), Chemistry Manager, and Safety/Fire Protection Coordinator. As such, the Plant Manager will report directly to the Site Executive Office, with the General Manager Support Services, the General Manager Operations and the General Manager Maintenance reporting directly to the PM. With the creation of the position of Chemistry Manager, the responsibilities for plant chemistry activities will be reassigned from the RES Manager to the Chemistry Manager. The Chemistry Manager will report directly to the General Manager Operations. The creation of the position of Safety/Fire Protection Coordinator removes the supervisory oversight responsibilities for the Fire Protection Program and the plant's Industrial Safety Program from the Security Manager and assigns it to the Safety/Fire Protection Coordinator. The Safety/Fire Protection Coordinator will report directly to the General Manager Support Services.

The changes are administrative in nature and do not involve plant equipment or plant operating conditions. The changes do not reduce the effectiveness of the management of activities or the oversight of plant operations. Therefore, these changes do not involve an unreviewed safety question.

JAF-SE-99-011, REV. 0: **Miscellaneous CAD System Discrepancies**

ENGINEERING CHANGE: **N/A**

The purpose of this safety evaluation is to evaluate the nuclear safety effects of various CAD system corrections to SAR documents.

The UFSAR changes are the following: change makeup nitrogen flow capacity from 30 scfm to 100 scfm; remove reference to buying bulk liquid nitrogen from a Syracuse, NY supplier; and to change the primary containment differential pressure system from the original designed configuration to the actual installed configuration. The CAD nitrogen system is designed to mitigate an accident that has already occurred. Changing the location of a bulk liquid nitrogen supplier from nearby Syracuse, NY to no reference for the location will still meet the three day time period requirement that is allowed for the delivery. Changing the nitrogen make-up flow capacity in the UFSAR from 30 scfm to 100 scfm will correct information regarding equipment capability, although no change in the actual design will occur since it remains 100 scfm. The primary containment differential pressure system was previously retired in place and is not required to be used in any accident mitigating role. The proposed activity does not involve an unreviewed safety question.

JAF-SE-99-012, REV. 1: **Plant Operation with 27AOV-118 Inoperable, Temp. Mod. To Disable 27AOV-118 and Temp Change to OP-37**

ENGINEERING CHANGE: **N/A**

Continued plant operation with 27AOV-118 inoperable and failed open, and 27AOV-117 closed and deactivated, as well as intermittent operation of 27AOV-117 under administrative control does not involve an Unreviewed Safety Question or place the plant in an unanalyzed condition. The NRC SER for Tech Spec Amendment 154 in March of 1990 approves operation in this manner. Tech Spec 3.7.D.2 and plant procedure ST-1H specifically address the controls required.

JAF-SE-99-014, REV. 0: RHR-Steam Condensing

ENGINEERING CHANGE: JD-99-099

RHR Steam Condensing Mode will be restricted to operation as directed by the Emergency Operating Procedures (EOPs). This ensures that 10MOV-21A/B, RHR Heat Exchanger Drain to Torus Isolation Valves, will be maintained shut to support Primary Containment Integrity until plant conditions are beyond the assumptions in the accident analysis. Steam Condensing Mode of RHR is not a safety-related function, and is not credited in the accident analysis or in the Appendix R Safe Shutdown Analysis. Therefore restricting this mode of RHR to ensure the components are aligned to support the safety-related function of Primary Containment Isolation does not constitute an unreviewed safety question.

JAF-SE-99-015, REV. 1: Modification to Eliminate RHR Header Pressurization by Opening 10SOV-263A and / or 10SOV-263B

ENGINEERING CHANGE: JD-99-085

The reason for this safety evaluation is to evaluate the use of 10SOV-263A/B, 10RHR-780A/B and/or 10SOV-264A/B for eliminating conditions associated with pressurizing the RHR pump discharge headers. It is proposed that these normally closed SOVs be allowed to be opened and the upstream manual valves throttled as necessary for the purpose of directing leakage into the Radwaste system, via sample panel 95SP-7 when leakage into the RHR system from a high pressure source causes the RHR header to become pressurized. The QA classification of 10SOV-263A/B solenoid operators will be revised (from QA Category I to QA Category II/III). These valves are currently in the IST program, but do not meet the criteria for inclusion based on there being no safety function for the valves to close. 10SOV-264A/B bodies and operators are classified QA category II/III and have no safety function.

The leakage contribution to RHR header pressurization is dependent on the rate of m-leakage from sources connected to the reactor coolant pressure boundary and the rate of out-leakage past the lines connected to containment or past the RHR pump discharge check valves. LLRT and independent test data collected using TST procedures confirm that leakage from containment isolation valves remains within allowable limits. The purpose of using sample lines from the RHR header to the sample sink in the Reactor Building is to allow this small amount of inleakage to be diverted to the Radwaste system where it can be processed.

This safety evaluation demonstrates that using the aforementioned valves in this manner does not constitute an unreviewed safety question since there are no adverse safety consequences associated with placing the valves in an open configuration for extended periods of time. Any leakage will not cause failure of any safety-related equipment, the dose associated with the leakage is bound by previous analyses and the design is consistent with 10CFR50, Appendix A, Criterion 55, Regulatory Guide 1.11 and ANS-56.2.

JAF-SE-99-016, REV. 0: Placement of Fire Damper 72FD-11 in the Closed Position

ENGINEERING CHANGE: JD-99-100

Fire damper 72FD-11, which is located in the wall separating the Administration Building from the Cable Spreading Room, will be administratively placed in the "closed" position in order to preclude significant CO_2 exfiltration from the Cable Spreading Room. This action is necessary since the design basis of the system credits closure of the subject damper in order to maintain the necessary concentration of CO_2 in the event of actuation of the Cable Spreading Room CO_2 fire suppression system. The closure of the fire damper will not adversely impact environmental conditions within the Cable Spreading Room during normal plant operating (non-fire) conditions. The evaluation demonstrates that the activity does not constitute any unreviewed safety question as defined by 10 CFR 50.59 mainly due to the fire damper being placed in its important to safety function.

JAF-SE-99-017, REV. 0: Permanent Removal of 66PCV-101

ENGINEERING CHANGE: JD-99-016

Evaluation conducted to support this NSE has determined that 66PCV-101 (RB Unit Coolers SWS Return Header Pressure Control Valve) has no function in the system, except as added balancing capability. This is not required since the flows through the unit coolers are balanced with the balancing valves on each individual cooler. Current ESW flow testing data per ST-8Q (Reference 2) indicates that there is limited margin on the service water flow to the Crescent Area Unit Coolers. This is mainly due to fouling of system piping and components with Microbiological Induced Corrosion (MIC). 66PCV-101 is a Masoneilan Camflex valve that offers some hydraulic resistance in the full open position. Replacing the failed open valve with a piping spool piece will slightly reduce unnecessary hydraulic resistance and result in an increase in emergency service water flow rate through the Crescent Area Unit Coolers.

This modification does not involve an unreviewed safety question, nor does it require a change to JAF Technical Specification.

JAF-SE-99-018, REV. 1: **NSE for EPIC System Upgrades: Temp. Mod. #99-047; M1-95-130; M1-90-212; F1-82-022**

ENGINEERING CHANGE: **N/A**

The proposed activities are: installation of a temporary network to the EPIC system; the installation of a new single switch/router (hub) on the EPIC output communication system, and the discrepancies associated with previously installed EPIC modifications as defined by USNRC Generic Letter, GL-95-02 for plant digital upgrades, The installation of temporary networks will be made by using approved temporary modifications and ensuring a firewall is installed between the EPIC system and the outside data communications circuit (LAN) to enforce one way communication from EPIC. This firewall will remain installed for system evaluation and then have its input disconnected, so that additional PC based systems can be evaluated for functionality using the next temporary modification.

There will be a slight increase in CPU utilization on the EPIC MASTER HOST for this temporary modification. The existing single switch/router (hub) is acceptable for use on the EPIC system as a team of network communication experts recommended it and NYPA approved. It thus, eliminated unacceptable network collisions which were occurring and causing computer communication failures, following the installation of DEC ALPHA computers with DEC VAX6520 computers. In order to minimize EPIC downtime, should the switch/router (hub) fail, a second switch/router (hub) will be purchased and staged in the EPIC computer room for immediate change over.

The temporary loss of the EPIC computer during power operation greater than 20% does not reduce the margin of safety since the RWM system is turned off. If the EPIC computer were unavailable during startup and/or reactor shutdown (≤10% power), the JAF Technical Specifications contain the requirements that must be met in order to change any rod positions other then SCRAM. If reactor power is between 10% and 20% of rated power and the RWM program is inoperative for any reason, the control room operators can maintain acceptable rod worth by simply adhering to the prescribed control rod patterns and sequences. Neither the EPIC system or RWM are required for the safe operation of the plant. The past installed modifications:

F1-82-022 "INSTALL EPIC SYSTEM"
M1-90-212 "UPGRADE EPIC HOST COMPUTERS"
M1-95-130 "UPGRADE EPIC NSSAA/NSSAB COMPUTERS"

And the discrepancies discovered while performing USNRC GL-95-02 reviews for digital plant equipment upgrades installed in JAF have been evaluated with operability determinations completed and the EPIC system is considered to be fully functional and acceptable for use by the JAF plant staff.

Based on the preceding evaluations, it was determined that the installation of a temporary network to the EPIC system, installation of a new single switch/router (hub) on the EPIC system and the discrepancies associated with previously installed EPIC modifications as defined by USNRC Generic Letter, GL-95-02 for plant digital upgrades, do not involve an unreviewed safety question as defined in 10 CFR50.59.

JAF-SE-99-020, REV. 0: 13MOV-15 Actuator Closing Logic

ENGINEERING CHANGE: JD-99-090

During Refuel Outage 13, Reactor Core Isolation Cooling (RCIC) Steam Supply Inboard Isolation Valve, 13MOV-15, failed its initial local leak rate test. The technicians performing the leak rate test found that shutting the valve using the manual handwheel significantly reduced the seat leakage. During the subsequent post outage review memorandum JMD-98-240 evaluated alternatives. It was determined based on industry experience and the vendor recommendation that the valve closing logic should be changed from torque control to limit switch control. Design change JD-99-090 provides the technical justification for this alternative.

Design Change JD-99-090, will modify the closing logic for 13MOV-15(OP). The logic change will place an existing spare position contact in series with the closing coil. The existing 2 and 2 rotor will be replaced with a qualified 3 and 1 rotor. The limit switch will be adjusted to allow the valve to reach the fully closed seat before it opens. This action will allow the valve to be closed to a precise point based on actual stem position rather then on torque.

This safety evaluation has shown that the installation and testing of design change JD-99-090 does not represent an unrevewied safety question.

JAF-SE-99-021, REV. 0: Installation of Permanent Temporary Power Jumpers

ENGINEERING CHANGE: JD-99-092

The purpose of this safety evaluation is to show that the installation, testing and associated procedure changes for design change JD-99-092, Installation of Permanent Temporary Power jumpers does not represent an unreviewed safety question. During Refuel Outage 13, temporary power jumpers were required to maintain equipment operating during scheduled bus outages. The equipment that received the temporary power was considered inoperable due to the lack of the normal power source. The jumpers did allow the equipment from the bus which would normally be de-energized to be energized and operating. The installation and removal of the jumpers during the refuel outage extended critical path time.

This design change will permanently install jumpers and allows operations to control the associated breakers and loading of the jumpers using an approved procedure. The jumpers will only be used during cold shutdown or refueling. The plant equipment, which is receiving temporary power, will not be considered operable. This design change is required to reduce critical path time during future refueling outages. Design Change JD-99-092 will permanently install jumpers between the following Motor Control Centers (MCC's):

- 71MCC-142 (QA Cat. II/III) supply to 71MCC-161 (QA Cat. I)
- 71MCC-143 (QA Cat. II/III) supply to 71MCC-162 (QA Cat. I)
- 71MCC-253 (QA Cat. I) to 71MCC-263 (QA Cat. I)
- 71MCC-133 (QA Cat. II/III) supply to 71MCC-152 (QA Cat. I)
- 71MCC-254 (QA Cat. I) to 71MCC-264 (QA Cat. I)

JAF-SE-99-022, REV. 0: **RHR Minimum Flow Bypass Valve Trip Setpoint and Setpoint Limit Change**

ENGINEERING CHANGE: **N/A**

This NSE evaluates replacing the FSAR stated RHR/LPCI Min Flow Trip Setpoint of "400 gpm" with the design basis minimum required flow for this function, "350 gpm per pump". FSAR Table 7.4-4 and section 7.4.3.5.4 will be revised to reflect the minimum required flow. The Field Trip Setpoint will be revised to reflect the calculated Limiting Trip Setpoint.

The increase in the Limiting Trip Setpoint reflects consideration of the design basis limit associated with the minimum flow required to prevent overheating the RHR pumps. Increasing the setpoint is conservative since this will ensure the minimum flow limit is not violated.

JAF-SE-99-023, REV. 0: Reactor Vessel Head Spray Connection Modification

ENGINEERING CHANGE: JD-99-087

Design Change JD-99-087 "Reactor Vessel Head Spray Connection Modification" will result in the permanent removal of the existing head spray nozzle assembly, including 6" X 4" reducing elbow and 4" blind flange, and the installation of a 6-inch ASME B16.5 1500 lb. blind flange (small tongue flange facing) and associated fasteners and gasket. The reactor vessel head spray connection forms part of the reactor vessel pressure boundary. To maintain the integrity of the reactor vessel pressure boundary (Safety Related function), an ASME B16.5 1500 lb. blind flange is specified which exceeds the reactor pressure vessel design pressure and temperature requirements of 1,250 psi gage and 575°F, respectively. In addition, the number of unisolable flanged connections within reactor vessel pressure boundary is reduced by one (1). Therefore, the Reactor Vessel Head Spray Connection Modification does not adversely impact any safety significant function. The changes proposed by this modification do not degrade the Security Plan, Quality Assurance Program, the Fire Protection Program, the Environmental Report or the Emergency Plan. This change does not constitute an Unreviewed Safety Question per 10 CFR 50.59.

JAF-SE-99-024, REV. 0: Containment Penetration X-18 Pressure Relief

ENGINEERING CHANGE: JD-99-093

Design Change JD-99-093 "Containment Penetration X-18 Pressure Relief' will result in the permanent installation of a pressure relief device (rupture disk assembly) in the Drywell Floor Drain Pump discharge piping (3"-WL-151-2) to provide pressure relief capability for Containment Penetration X-18. The addition of a rupture disk precludes thermally induced overpressurization of isolated, water-filled piping sections that penetrate containment which could jeopardize the ability of accident-mitigating systems to perform their safety functions and, consequently, lead to a breach of containment integrity via bypass leakage. This design change addresses the issues discussed in Nuclear Regulatory Commission Generic Letter (GL) 96-06. The rupture disk has been designed to ensure rupture occurs at a pressure greater than the hydrostatic test pressure of the piping and below the maximum allowable working pressure for Class 151 piping under DBA-LOCA conditions. Therefore, the Containment Penetration X-18 Pressure Relief design change does not adversely impact any safety significant function.

JAF-SE-99-026, REV. 0: **Installation of a By-Pass Line Around Oil/Water Separator (20SP-14)**

ENGINEERING CHANGE: **JD-97-054**

Design Change JD-97-054, "Installation of a By-pass Line Around Oil/Water Separator (20SP-14)" will install a by-pass line (3"-WL-163-812) including a 3-inch plug valve 20RDW-270 (Oil/Water Separator SP-14 Bypass Manual Flow Control Valve), around the oil/water separator (20SP-14). The by-pass line (3"-WL-163-812) will include a 1½-inch branch connection, with a 1½-inch gate valve 20RDW-271 (Oil/Water Separator SP-14 Flush/Test Valve). Implementation of this design change will eliminate the existing crud trap within oil/water separator (20SP-14) which results in area high radiation and personnel dose in the Radwaste Pump Room (EL. 250'-0"). While the Radwaste System is classified as a nuclear safety-related system (the Safety Related functions of the Radwaste System and associated components are to help maintain integrity of primary containment boundary and to provide containment isolation valve position indication), the change in oil/water separator (20SP-14) piping arrangement has no impact on any safety significant function. This change does not constitute an Unreviewed Safety Question per 10 CFR 50.59.

JAF-SE-99-027, REV. 1: **Digital Upgrades to the H2/O2 Containment Atmosphere Monitoring (CAM) System**

ENGINEERING CHANGE: **JD-99-017**

This evaluation determines the acceptability of the current as-built configuration of the H2/O2 Containment Atmosphere Monitoring System (CAMS) based on the criteria provided by NRC Generic Letter 95-02, Use of NUMARC / EPRI Report TR-102348, "Guideline on Licensing Digital Upgrades", in Determining the Acceptability of Performing Analog-to-Digital Replacements Under 10 CFR 50.59. Secondly, the evaluation addresses proposed changes to the FSAR that describe the current configuration of H2/O2 CAMS and corrects inconsistencies between the descriptions in chapters 5.2.3 and 14.6.2 of the FSAR as well as editorial errors in Tables 7.3-6 and 7.19-1. Finally, the evaluation determines the impact of microprocessor hardware and software (firmware) changes per Design Change JD-99-017 on the safe and reliable operation of the CAM system. The evaluation performed concludes that these changes do not create an unreviewed safety question.

JAF-SE-99-028, REV. 0: **Evaluation of Main Steam and Offgas NUMAC Log Radiation Monitors to Generic Letter 95-02 Criteria**

ENGINEERING CHANGE: **N/A**

This evaluation determines the acceptability of the current as-built configuration of the Main Steam and Off-gas NUMAC Log Radiation Monitoring Systems based on the criteria provided by NRC Generic Letter 95-02, Use of NUMARC / EPRI Report TR-102348, "Guideline on Licensing Digital Upgrades", in Determining the Acceptability of Performing Analog-to-Digital Replacements Under 10 CFR 50.59. Secondly, the evaluation addresses an editorial correction to FSAR section 7.12.1 which describes the current configuration of the Main Steam NUMAC Log Radiation Monitors as having three trips when there are actually four. The evaluation performed concludes that these changes do not create an unreviewed safety question.

JAF-SE-99-030, REV. 0: Evaluation of Control Rod Fast Withdrawal Velocity

ENGINEERING CHANGE: N/A

During performance of ST-20T, Post Scram Control Rod Timing, rod drive withdrawal times were sometimes faster than the specified time of 28.8 seconds, which corresponds to the maximum previously evaluated withdrawal speed of 5 inches per second specified in UFSAR Section 3.5.6.1.

GE and NYPA evaluations of the effects of withdrawal speeds up to 5.7 inches per second conclude that the CRD safety function of rapid insertion during a plant scram is not affected by the fast withdrawal speed, and that the impact of the higher withdrawal speed on analyzed transients involving rod withdrawal remains within the specified limits for those transient analyses. This evaluation applies to FitzPatrick Cycle 14 operations only.

The fast withdrawal speed is the result of internal leakage in the control rod drive mechanism. While some sources of internal leakage adversely affect drive operation, multiple failures are required including the leakage to cause an uncontrolled rod withdrawal. The consequences of such failures are bounded by the design basis Control Rod Drop Accident described in UFSAR Section 14.6.1.2; since the observed withdrawal speed does not approach the assumed rod drop speed, the consequences of such an accident are not increased. The leakage may also adversely affect individual rod scram times. Since the scram time requirements of Technical Specification 3.3.C continue to be met, this neither increases the probability of a failure of the scram function nor adversely affects the consequences of accidents, transients, or equipment failures for which a scram is required.

Accordingly, operation with control rod withdrawal speeds up to 5.7 inches per second does not create an unreviewed safety question.

JAF-SE-99-032, REV. 0: Alternate Position Indicating Probes (PIP's)

ENGINEERING CHANGE: JD-99-122

The use of the alternate / original position indicating probes has been evaluated, within design change JD-99-122, and found to be acceptable for use at the JAF facility. The existing description of the control rod drive components within the FSAR indicates that each position switch on the position indicating probe is sheathed in a braided glass sleeve. This sleeve is intended to provide a media to aid in the mounting / protection of the reed switch and to provide for reed switch terminal insulation. The alternate control drive position indicating probes do not employ the use of a braided glass sleeve on the position switches. The replacement position indicating probes, General Electric part number 797E111G001, do not use the fiberglass sheathing over the switches; however those design issues for which the fiberglass sheathing was originally installed is accomplished by:

- Suspending the reed switches away from contact with the switch support by mounting them on insulating pins affixed to a strip assembly; with the spring clip eliminated, a focal point for the transmission of vibration and impact forces to the reed switch is also eliminated,
- Terminating the switch connections to the insulating pins which eliminate the potential contact between the end connectors of the reed switches.

Therefore, the use of the replacement PIP does not impact the operation of the control rod drive or reactor manual control system / control rod drive position indicating switch indications. As a result of this evaluation there exist no Unreviewed Safety Questions (USQ).

JAF-SE-99-033, REV. 0: MSIV Enhancements

ENGINEERING CHANGE: JD-99-089

Design Change JD-99-089 "Main Steam Isolation Valve (MSIV) Enhancements" will allow optional maintenance activities (MSIV Enhancements) associated with the Main Steam Isolation Valves (MSIVs) (29AOV-80A, -80B, -80C, -80D, -86A, -86B, -86C, -86D) to ensure acceptable valve performance. A listing of the specific enhancement options is as follows:

- Installation of Hardened Washers Between Body/Bonnet Nut and Bonnet to Reduce Torque Transfer Losses and Eliminate Bonnet Leakage
- Installation of Installation of Hardened Washer at Both Ends of Live Load Springs and Nut and Packing Gland to Reduce Torque Losses for Added Packing Pressure to Better Control Packing Leakage
- Installation of Junk Ring / Spacers in Place of Carbon Bushing in Packing Gland to Reduce Stem Side Loading
- Installation of Articulating Coupling to Minimize Actuator Misalignment
Installation of Improved Body Guiding Pads for Improved Disc to Seat Alignment

The MSIV enhancements are intended to reduce required maintenance and to decrease Local Leak Rate Test (LLRT) failures. As such, the MSIV Enhancement design change Does not adversely impact any safety significant function. The changes proposed by this Design change do not degrade the Security Plan, Quality Assurance Program, the Fire Protection Program, the Environmental Report or the Emergency Plan. This change Does not constitute an Unreviewed Safety Question per 10 CFR 50.59.

JAF-SE-99-034, REV. 0: RPV Head Strongback and Carousel Assembly Upgrade

ENGINEERING CHANGE: JD-98-153

Design Change JD-98-153 proposes the use of a GE provided strongback/carousel assembly, upgraded stud tensioners, and a single nut/washer storage rack for disassembling, rigging, staging, and reassembling the RPV head during refueling outages. Use of this equipment will allow for a more efficient disassembly and reassembly process, thereby reducing time and manpower requirements. Uncontrolled drops of the new strongback /carousel are not considered credible since it was demonstrated that NUREG-0612 load handling requirements will be satisfied via qualification of the hardware and procedural restrictions. The probability of the RPV head drop and associated consequences have not been increased from that previously analyzed. In addition, storage of the hardware on the refueling floor between outages will have no adverse impact on the safety or operation of the plant.

In summary, it has been demonstrated that implementation of Design Change JD-98-153 does not to constitute an unreviewed safety question as defined by 10CFR 50.59. Therefore, the activity can be performed as proposed.

JAF-SE-99-035, REV. 0: Procedure Periodic Reviews

ENGINEERING CHANGE: N/A

Section 6.8(A) of Tech Specs Vol. 1A and Section 7.2 of Tech Specs Vol. 1B commit JAF to meet or exceed the requirements of Section 5 of ANSI N18.7-1972 and Appendix A of Regulatory Guide 1.33, November 1972. Neither of these documents specify a required frequency for procedure reviews. However, ANSI N18.7-1972 places the responsibility of specifying the frequency of such reviews upon the Licensee, and states:

Each procedure shall be reviewed prior to initial use and periodically thereafter. The frequency of such reviews shall be specified.

The purpose of this Safety Evaluation is to define the frequency of procedure periodic reviews for Technical procedures as CONTINUOUS, and to eliminate the administrative requirement to perform 5 year periodic review of Technical procedures. The process of maintaining procedures current and accurate is already in place at JAF. Eliminating the requirement to perform a procedure review at a specified time interval will have no impact upon the safe and reliable operation of the facility, and does not constitute an Unreviewed Safety Question.

JAF-SE-99-036, REV. 0: Revision of FSAR Section 9.8.3.2 Carbon Dioxide Systems

ENGINEERING CHANGE: N/A

FSAR section 9.8.3.2, Carbon Dioxide Systems, will be revised to reflect the current interface configuration between the plant's total flooding carbon dioxide fire suppression and ventilation systems. The statement "Exhaust and supply fans in CO_2 protected areas will be automatically shut down upon actuation of that area's CO_2 system" will be deleted and additional description of the current ventilation interfaces will be added.

The ability of the plant's CO_2 fire extinguishing systems to achieve and maintain the minimum required CO_2 concentration is based upon tests conducted in 1994. The tests assure that the current ventilation system interfaces do not adversely impact CO_2 system performance. Additionally, the tests and subsequent calculations identified the potential for overpressurization of several CO_2 protected enclosures during system discharge and consequently systems were modified to alleviate the potential.

The activity does not constitute an unreviewed safety question nor does it adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

JAF-SE-00-001, REV. 0: 12MOV-15 Actuator Closing Logic

ENGINEERING CHANGE: JD-99-163

The Reactor Water Cleanup (RWCU) Inboard Isolation Valve has a history of local leak rate failures. DER 97-00527 documents the failure history of the valve. It was determined based on industry experience and the vendor recommendation that the valve closing logic should be changed from torque control to limit switch control. Design change JD-99-163 provides the technical justification for this alternative.

Design Change JD-99-163, will modify the closing logic for 12MOV-15(OP). The logic change will place a spare position contact in series with the closing coil (ESK-6MAF, Revision 16A-0, Elementary Diagram 600V. CKTS-MOV RHR Shutdown Cool. And RWCU Sys. Isolation Valves). The existing 2 and 2 rotor will be replaced with a qualified 3 and 1 rotor. The limit switch will be adjusted to open and stop the MOV when the closed seat is contacted. The limit switch adjustment will prevent the additional torque from being applied when closure is based on torque switch setting. The additional torque is causing the seating surfaces to deform and leak. This action will allow the valve to be closed to a precise point based on actual stem position rather than on torque.

This safety evaluation has shown that the installation and testing of design change JD-99-163 does not represent an unreviewed safety question.

JAF-SE-00-002, REV. 0: Change to Section 17.2 of JAFNPP FSAR

ENGINEERING CHANGE: N/A

The changes proposed by this Nuclear Safety Evaluation involve the elimination of the Cross Reference Matrix of QA Implementing Procedures in the FSAR.

The changes are administrative in nature and do not involve plant equipment or plant operating conditions. The changes do not reduce the effectiveness of the QA Program, management of activities, or the oversight of plant operations. Therefore, these changes do not involve an unreviewed safety question.

JAF-SE-00-004, REV. 0: Spent Fuel Pool Cleanup

ENGINEERING CHANGE: N/A

This activity evaluates the lifting of heavy loads associated with a cleanup of the JAF Spent Fuel Pool, including shipping cask, liners, transfer bell, chrusher-shear machine and other tools and equipment. In addition, it evaluates the temporary bypassing of the zone interlocks of the Reactor Building Crane, so that the 20 Ton and 125 Ton hoists can be used in a larger portion of the cask area of the spent fuel pool (SFP).

All heavy load lifts during the proposed activity will be accomplished using the single-failure-proof hoists of the Reactor Building Crane, and the guidelines contained within NUREG-0612. This will include the defense-in- depth principles of the heavy load program and a minimum safety factor of 10:1 on ultimate strength for rigging equipment. As such, the loads are not postulated to drop, i.e., the probability of load drop is extremely low, which meets the single failure criteria of NUREG-0612.

Bypassing of the SFP zone interlocks has also been determined to be acceptable. Specifically, administrative controls to maintain sufficient clearance, combined with safe load paths and empty rack cells along the north edge of the cask area will ensure that there is no inadvertent damage to any spent fuel assemblies or the spent fuel pool liner.

This evaluation demonstrates that this activity using the Reactor Building Crane does not constitute an unreviewed safety question as defined by 10 CFR 50.59, nor does it involve significant hazards defined in 10 CFR 50.92. Therefore, the activity can be performed as proposed.

JAF-SE-00-005, REV. 1: Organizational Changes – Nuclear Generation Realignment

ENGINEERING CHANGE: N/A

The purpose of this NSE is to describe and evaluate proposed changes to the NYPA organizational structure, as described in the JAF FSAR. These changes involve the realignment of the Nuclear Generation department and the resulting changes in reporting relationships. These changes include:

- The removal of the position of Vice President – Appraisal and Compliance (VP-A&CS) from the nuclear organization and the resulting changes in reporting relationships; and
- The removal of the position of Executive Vice President – Project Operations (EVP-PO) from the nuclear organization and the resulting changes in reporting relationships;
- The revision of the position title of Vice President – Engineering to Vice President – Engineering and Project Controls; and
- The transfer of responsibility for the Nuclear Safety Speakout Program from the Director – Security to the Director – Regulatory Affairs and Special Projects.

These changes do not eliminate any functional requirements. The proposed changes are administrative in nature and do not involve plant equipment or operating conditions. They will not reduce the effectiveness of the management of activities or of the oversight of plant operations. Therefore, these changes do not involve an unreviewed safety question.

JAF-SE-00-006, REV. 0: Online Adjustment of Recirculation MG Set High Speed Stops

ENGINEERING CHANGE: N/A

Scoop Tube Positioner mechanical stops and other hardware limits are used to limit the maximum attainable Recirculation flow in support of assumptions in the COLR. However, following calibration of total core flow instrumentation, these limits may prevent attaining rated core flow. It is therefore desirable to adjust these stops during plant operation.

The methods of adjusting the stops are not addressed in the SAR. During all adjustments and testing other than the adjustment of the mechanical stops themselves, the mechanical stops are in force at or below the required setpoints and provide the required protection.

During adjustment of the mechanical stops, the Scoop Tube Positioner is operated in manual mode, which precludes transients initiated by failure of the Recirculation Flow Control instrumentation. Operating and adjustment procedures contain operational limits that ensure that core flow does not approach the value assumed in the COLR. In the unlikely event of failure or inadvertent release of the manual handcrank, Recirculation Pump speed would decrease at a rate somewhat less than the normal controlled rate; this transient would be well within the bounds of flow decrease transients analyzed in the SAR.

The transients or malfunctions that could arise during adjustment of the mechanical stops are therefore bounded by the Recirculation flow transients analyzed in the SAR, and no new transients or failure modes are created. The COLR limits are maintained during mechanical stop adjustment because manual operation of the Scoop Tube Positioner precludes a failure-induced increase in Recirculation flow rate, and procedural limits ensure that operators do not inadvertently exceed the assumed maximum flow. Therefore, online adjustment of the Scoop Tube Positioner high-speed stops does not involve any unreviewed safety question.

JAF-SE-00-008, REV. 0: Construction of Independent Spent Fuel Storage Installation

ENGINEERING CHANGE: F1-91-103

The purpose of the proposed activity is to construct an Independent Spent Fuel Storage Installation (ISFSI) to support future storage of JAFNPP spent nuclear fuel assemblies, in NRC approved storage casks. The JAF ISFSI will employ the general license conferred in 10 CFR Part 72. The scope of this activity consists of installation and acceptance testing of the JAF ISFSI structures, systems, and components that will be located in the JAF Yard, including the ISFSI storage pad, the cask assembly pad, the roadway between the ISFSI storage pad and the cask assembly pad, the ISFSI area nuisance fence, the ISFSI grounding system, the ISFSI temperature monitoring system, the ISFSI electrical supply, JAF Security System modifications, JAF area lighting modifications, and ISFSI area grading and drainage. The scope of this activity does not include deployment of the spent fuel storage casks.

The construction and testing of the JAF ISFSI structures, systems and components under this activity will not affect the function or operation of any JAFNPP structure, system or component important to safety. The JAF ISFSI structures are completely passive, and are located away from and do not interface with JAFNPP SSCs important to safety. The JAF ISFSI electrical and instrument systems do not interface with any JAFNPP systems or components important to safety.

JAF ISFSI construction activities and the JAF ISFSI post construction condition will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the UFSAR, will not create the possibility of an accident or malfunction of equipment important to safety of a different type as previously evaluated in the UFSAR, and will not reduce the margin of safety as defined in the basis of any Technical Specification. Therefore, the proposed activity does not involve an unreviewed safety question. No changes to the JAFNPP Technical Specifications are required to support the proposed activity.

The proposed changes to the JAF Security System include installation of a new protected area gate, and utilization of an existing low voltage power supply. The proposed changes to the JAF area lighting system includes relocation and upgrade of an existing area light pole. These proposed changes have been designed to meet JAF Security Program commitments, and therefore, will not degrade the safeguards effectiveness of the JAF Security Plans. The proposed activity will not have any adverse affect on the Quality Assurance Program, the Fire Protection Program, or the Emergency Plan. There are no significant unreviewed environmental impacts associated with this activity.

JAF-SE-00-009, REV. 0: **Primary Containment Atmosphere Control and Dilution (Nitrogen) System Materials Discrepancy**

ENGINEERING CHANGE: **N/A**

The purpose of this safety evaluation is to revise section 5.2.3.8.10 of the UFSAR. This section implies that stainless steel materials are required for the low temperature portion of the nitrogen system. Other materials currently exist in the low temperature nitrogen system as described by DER 00-00228. Other errors and omission were found including: 1) 20°F value; 2) use of the word "all"; 3) some stainless steels are brittle; 4) material downstream of electric heaters is not limited to carbon steel pipe; and 5) discussion of embrittlement (omission).

This NSE supports an FSAR change package for modifying section 5.2.3.8.10 to correct the errors and one omission. Errors and omission found include: 1) stainless steel only requirement for low temperature applications; 2) 20°F value; 3) use of the word "all"; 4) some stainless steels are brittle; 5) material downstream of electric heaters can have more than just carbon steel pipe; and 6) discussion of embrittlement (omission). The stainless steel only "literal" statement was really a design requirement to use materials that would not be subject to embrittlement at low temperatures. The 20°F was changed to -20°F primarily based on design basis information in APO-90. The word "all" was not appropriate since the design bases (APO-90) showed that materials other than stainless steel were intended to be used. The word "austenitic" was added to "stainless steels", since most other stainless steels are brittle at cryogenic temperatures and APO-90 used all austenitic stainless steels. The piping and components downstream of the ambient vaporizer and electric heaters are not necessarily designed for subzero or cryogenic temperatures regardless of whether it is carbon steel or another material. The ambient vaporizers and the electric heaters have an important function to protect the downstream piping and components from brittle fracture/failure.

JAF-SE-00-010, REV. 0: JAF Fire Protection Plan, Revision 6

ENGINEERING CHANGE: N/A

The Fire Protection Plan (FPP) will be revised to delete detailed responsibility statements for each position supporting the fire protection program. The detailed responsibility statements will be relocated to a site level procedure and the responsibilities within the FPP will be general in nature as appropriate for a higher level program document. The FPP will also be revised to reflect organizational changes including the deletion of the positions of Fire Protection Supervisor and the Fire Protection Program Engineer which were combined into the newly created position of Fire Protection/Safety Coordinator. Additionally, professional qualifications will be added for the Fire Protection/Safety Coordinator, Fire Protection System Engineer, Fire/Safety Specialists and the Training Specialist responsible for fire brigade training.

The FPP does not control the operation or configuration of plant systems, equipment, or components, nor does it provide guidance in achieving and maintaining safe shutdown in the event of a fire.

Based on the above, the proposed revision to the FPP does not constitute an unreviewed safety question and does not adversely impact the ability to achieve and maintain safe shutdown in the event of a fire.

JAF-SE-00-014, REV. 2: SRV Electric Lift/ATWS Setpoint Change Project

ENGINEERING CHANGE: M1-97-070

Modification M1-97-070:

1. Installs an SRV Electric Lift System that adds electric actuation to the SRV which provides a diversified means of opening the SRVs to improve valve opening reliability (by overcoming the effects of corrosion-induced bonding between the pilot disc and the pilot seat).

2. Modifies the reactor level setpoint associated with ATWS Level Recirculation Pump trip and ARI.

3. Installs 7 new permanent test connections on the instrument gas supply tubing for the 7 ADS SRVs.

This modification does not involve an unreviewed safety question. However, revisions will be required to the FSAR and the Technical Specifications Bases. There is a change required to the JAF Technical specifications, but it is evaluated separately from this safety evaluation (JPN-00-002, dated February 3, 2000 "Request for Technical Specifications Change regarding Automatic Transients without Scram RPT/ARI Setpoint Change" (JPTS-99-008)).

JAF-SE-00-015, REV. 0: Cleanup Phase Separator Level Instrument Replacement

ENGINEERING CHANGE: JD-97-090

Design Change JD-97-090 will replace the existing Phase Separator Tank Level Instruments. The existing bubbler level instruments and sludge point sensors are no longer operable and require replacement. The design change will also restore interlocks for the Mixing Pump (20P-43) and Decant Pump (20P-44) control circuits which have not been functional since the level instruments and the centrifuge were taken out of service. Procedure changes will be made to OP-48 and OP-49 to provide direction for system operation with the modified control circuits. This NSE evaluates the impact of installation, testing, turnover and operation of the Phase Separator Tank Level Instrument Replacement Design Change, JD-97-090.

JAF-SE-00-016, REV. 0: **EPIC Rods-In Monitor and Control Rod Scram Timing Program Upgrade**

ENGINEERING CHANGE: **JD-99-020**

Design change JD-99-020 will install a Rods In Monitoring Program on the EPIC system. The existing Full In indication relies on the position indicating probe reed switches for position 00. The program will use an output from the RPIS Probe Buffer Cards which will provide rod movement data to EPIC. The movement data uses an unlatched rod drift signal. This will not rely on the single position switch. Following a SCRAM the rod movement and the initial rod position data will be used by the program to calculate the final rod positions. The calculated rod positions will be provided to the operators as a display on the EPIC monitors in the Control Room. This will provide the operators another method of determining that the reactor is shutdown and prevent unnecessary entry into EOP-3.

Design change JD-99-020 will also add a scram time testing program to EPIC. The scram time testing program will provide automatic scram timing of all fully withdrawn control rods following a full reactor SCRAM. The program will also provide a new method for individual scram timing. The scram timing method will meet the requirements specified in the proposed Improved Technical Specifications (ITS) Section 3.1.4. This program will not be used to meet the requirements specified in the current Technical Specifications.

This NSE evaluates the impact of equipment failures, installation, testing, partial/complete turnover and operation of the EPIC Rods In Monitoring Program and Control Rod Scram timing Program.

JAF-SE-00-017, REV. 0: Eliminate RBCLC Drywell Temperature Alarm

ENGINEERING CHANGE: N/A

Table 4.10-1 in the UFSAR indicates there is an alarm associated with the drywell coolers closed cooling water (CCW), and no such alarm exists. The reference to this alarm will be removed. Also in section 4.10, there is a paragraph that discusses the CCW differential temperature between the inlet and outlet of the Air/Nitrogen coolers. This paragraph will be changed to include a statement that says "even though a rise in the differential temperature could be an indication of a leak, there is no alarm associated with this reading due to the difficulty of detecting a small leak in this manner and there currently are more accurate and timely ways of detecting such a leak."

The above changes will not create an unreviewed safety concern, currently in the UFSAR in section 4.10, under the heading of "Detection of Leakage Inside Drywell" it states "as described in Technical Specifications there are two Leak Detection Systems used for monitoring the identified and unidentified leakage's: (1) the drywell sump monitoring system which monitors the floor drain sump and the equipment drain sump; (2) the drywell continuous atmosphere radioactivity monitoring system which has three channels, each of which monitors particulate, iodine and noble gases." There is no reference to an alarm associated with the Drywell Coolers CCW temperature. This section continues to state 9 monitoring systems or sub systems, none of which mention the Drywell Coolers CCW temperature. And in Section 7.3.4.2 titled "Primary Containment Leak Detection System Instrumentation" it is states the features of this system which monitor for leakage inside and outside the primary containment. Again there is no mention to the Drywell Coolers CCW temperature.

Any leak in PC will be detected at its respective limit as described in the above referenced sections of the SAR. The ability for the CCW to detect a 5 gpm leak in PC is very limited to due several factors. This is due to the small effect that a 5 gpm steam leak would have on such a large volume of Air/Nitrogen, and the corresponding small rise in CCW water temperature to this increase. The ability to see this small of a leak is also dependent on the leaks location, as much of the Air/Nitrogen is returned to the Drywell Coolers from the top of the Drywell. A leak at the lower portion of the Drywell would be very hard to identify due to the fact the leak would have a smaller effect on the return Air/Nitrogen temperature. Other factors that would make such a small leak hard to identify, are changes in RBCLC flow rate, lake temperature and changing heat exchange efficiencies.

JAF-SE-00-018, REV. 0: Replace 12MOV-18 and Feeder Breaker for 12MOV-18

ENGINEERING CHANGE: JD-99-134

The existing Reactor Water Cleanup (RWCU) System Pump Suction Primary Containment Inboard Isolation Valve (12MOV-18) has been susceptible to failure of10CFR50, Appendix J, Type C local leak rate tests performed in accordance with the James A. Fitzpatrick Primary Containment Leakage Rate Testing Program.

The existing gate valve is being replaced with a gate valve of a different style. The pressure locking vent valve is being removed, as there is no potential for pressure locking. An additional Local Leak Rate Test connection is being installed on the bonnet of the valve, which can accommodate an alternate test methodology. The existing motor operator will be reinstalled on the replacement valve. Although currently acceptable, the breaker size is being increased to improve the electrical reliability of the valve. The purpose of this evaluation is to show that the proposed activities do not present an unreviewed safety question. The new valve does not change the functions and control scheme for this primary containment isolation valve.

The new valve is expected to perform better and provide additional safety margin during accidents and transients. No new failure modes are introduced by this design and design changes have no adverse impact on existing equipment, which would prevent the equipment from performing their required functions. The associated design basis requirements and Technical Specifications are unaffected and are satisfied by the new valve design and installation. Therefore, the replacement of the 12MOV-18 valve does not adversely affect the operational capability of the valve, compromise the safe operation of the plant or introduce an unreviewed safety question.

JAF-SE-00-020, REV. 0: System 10, 10MOV-26A/B Actuator Replacement

ENGINEERING CHANGE: JD-99-088

Modification JD-99-088 will replace the existing SB-00-15 Limitorque actuators with larger capacity SB-0-15 actuators for valves 10MOV-26A/B. The SB-0-15 actuator is designed with a greater load rating than the SB-00 actuator. The modification activity was proposed to alleviate valve actuation cycle tracking imposed against these valves due to the MOV Program required torque setting, which exceeds Limitorque standard actuator ratings. The actuation cycle limitations were imposed on the basis of a fatigue life calculation.

The modified valves/actuators do not change the functions and control scheme of the RHR containment spray outboard isolation valves. The modification is expected to provide better performance and reliability for the operation of the valves. With the larger actuator, additional design margin has been gained. No new failure modes are introduced by these design changes. The associated design basis requirements and Technical Specifications are unaffected and are satisfied by the new valve/actuator design and installation. Therefore, the modification of the valves/actuators for 10MOV-26A/ B does not adversely affect the operational capability of the valve or the containment spray portion of the RHR system, compromise the safe operation of the plant, or introduce an unreviewed safety question.

JAF-SE-00-021, REV. 0: **FSAR Open Item 883 resolution, Analysis Regarding LOOP While Venting or Purging Containment to SGTS**

ENGINEERING CHANGE: **N/A**

The purpose of this safety evaluation is to evaluate the clarification of the design basis accident applicable to the SBGT Survivability calculation as documented in FSAR section 5.3.4. This evaluation will determine the impact of changing the FSAR as delineated below, on nuclear safety and will determine if the change will result in an unreviewed safety question.

FSAR section 5.3.4, page 5.3-9, 2nd sentence of 4th paragraph will change from: "In the event of a design basis LOCA, the maximum flow through this 6 inch line will not exceed the design capabilities of the Standby Gas Treatment System."

It will change to: "In the event of a LOCA without Loss Of Offsite Power (LOOP), the maximum flow through this 6 inch line will not exceed the design capabilities of the Standby Gas Treatment System."

Technical Specification Bases section 3.7.C, last paragraph on page 191 will change as follows: The first and last sentence contain the statement "design basis LOCA", will be changed to "LOCA without Loss Of Offsite Power (LOOP)".

There will be no changes to the primary containment system or its interfaces. There will be no changes to the standby gas treatment system or its interfaces. What is changing is the type of LOCA applicable to SBGT Survivability analysis when purging and venting containment to the standby gas treatment system during the postulated accident. NYPA licensing reviewed the licensing/design basis documentation regarding containment vent and purge during normal plant operation and documented the review on memo JLIC-98-050. NYPA licensing concurred with Stone & Webster's position that for this analysis a LOOP was not considered and is not required. 'There will be no change to any Containment allowable values, as specified in the Technical Specifications, resulting from this proposed change. Changing the FSAR and the Technical Specification Bases to reflect the actual licensing basis has not resulted in an unreviewed safety question.

JAF-SE-00-022, REV. 0: Auxiliary Work Bridge Improvements

ENGINEERING CHANGE: JD-00-029

The purpose of this safety evaluation is to show that the issues associated with modifying and using the reconfigured "Auxiliary Work Bridge" on the refueling floor does not represent an unreviewed safety question. The use of the "Auxiliary Work Bridge" for use in normal in-vessel activities and to allow for parallel activities while the refueling bridge is in use, as described in this evaluation. does not adversely affect any structure, system or component (SSC) assumed to operate during normal plant operation or an accident. There are no physical or functional changes in any structure, system, or component, or changes to the design criteria as a result of this activity. Changes evaluated by this NSE are consistent with original design requirements. The evaluation demonstrates that the activity does not constitute any unreviewed safety question as defined by 10 CFR 50.59 nor does it involve significant hazards as defined in 10 CFR 50.92. Therefore, the activity can be performed as proposed.

JAF-SE-00-023, REV. 0: **ASTM E71 Superseded By ASTM E446 Regarding Acceptability of Radiographs For Steel Castings**

ENGINEERING CHANGE: **N/A**

The proposed change to Section 16.5.5.1.1 of the FSAR to include the new ASTM Standard E446, which has replaced the presently referenced ASTM Standard E71, for providing example radiographs to be used in evaluating the radiographs of steel castings will not adversely affect plant safety due to the fact that the newer standard provides for a more conservative evaluation than the older standard. The change to add the newer standard simply updates the FSAR to align with the newer standard number used in other related codes and standards.

JAF-SE-00-024, REV. 0: Emergency Service Water Hydraulic Model Benchmark Testing

ENGINEERING CHANGE: N/A

Temporary pressure gauges will be installed on Emergency Service Water, Control Room/Relay Room cooling and Reactor Building Closed Loop Cooling branch lines to support benchmarking of a hydraulic model for the Emergency Service Water System. The impact on safety system pressure boundary integrity and primary containment integrity are evaluated for the temporary plant configuration.

JAF-SE-00-025, REV. 3: **Provide Chemical Cleaning Process for ESW Piping and Heat Exchangers**

ENGINEERING CHANGE: **N/A**

The Emergency Service Water (ESW) has had reduced flows to ESW unit coolers and heat exchangers as the result of fouling due to a combination of silt and iron oxide accumulating in the piping and heat exchangers. Chemical cleaning of the ESW piping and heat exchangers will be conducted to improve and maintain heat exchanger heat removal capability operability margins. The chemicals used in the chemical cleaning process are not detrimental to the internal surfaces of the ESW system. The cleaning and neutralizing solution will be collected and processed in accordance with AP-9.11. The cleaning process will be monitored by Chemistry for isolation valve leakage and for appropriate chemical concentration levels in the cleaning and neutralizing solutions. While the cleaning process is being performed the piping and or heat exchanger will be out of service. An LCO will be entered, as required, or in the case of the crescent unit coolers an LCO is not required, as only 4 of 5 crescent unit coolers are required to remove total accident heat loads.

The pumping skid and equipment were chosen for their resistance to chemicals. The chemical cleaning process is performed at pressures below that of the ESW design pressure. Precautions have been taken to preclude the effects of chemical spillage or spray. The pumping / chemical cleaning skid will have 24 hour coverage.

The appropriate combustible loading and penetration fire breech will be controlled by approved procedures. Future cleaning process applications will be addressed either by TOP or WR step text. The location of the skid and floor loading will be controlled by approved procedures.

The chemical cleaning of the ESW piping and heat exchangers only enhances ESW system performance and does not involve an Unreviewed Safety Question or place the plant in an unanalyzed condition.

JAF-SE-00-026, REV. 0: **Change to Section 7.1.14 of JAF FSAR – Remove Reference to Diesel Fuel Oil Line Flow Meter Calibration**

ENGINEERING CHANGE: **N/A**

The proposed change involves the elimination of any reference to the emergency diesel generator fuel oil line flow meter check and calibration in the FSAR.

Since the flow meters provide local indication only and do not provide any active safety related function, they do not meet the criteria established in AEC Question 7.3, Item 10 for "engineered safety feature instrumentation". Therefore, it should not have been included in the 1972 NYPA response. Further promulgating the error, the 1982 FSAR update that incorporated the AEC questions and responses erroneously included the implied commitment to calibrate the flow meters. The flow meters do not provide any control or actuation signal and, aside from their passive system integrity function, are provided solely as a means to assess diesel fuel oil transfer pump capability during pump surveillance testing. There is no requirement to calibrate or check the flow meters, therefore, removing its reference from the FSAR does not involve an unreviewed safety question.

JAF-SE-00-027, REV. 0: Nuclear Engineering Organization Changes

ENGINEERING CHANGE: N/A

The organizational changes will not reduce the effectiveness of the Nuclear Engineering organization and will not reduce oversight of plant operations. The changes to the Engineering Division are organizational in nature.

The title of the JAF Director Design Engineering is changed to the JAF Director of Engineering, the JAF Director of Engineering will report directly to the Vice President - Engineering and Project Controls. The title of the Manager Project Engineering is changed to the Manager Project Management who will now report to the JAF Director of Engineering.

The Technical Services organization will now report to the JAF Director of Engineering instead of the General Manager Operations. The positions of Manager Mechanical/Civil Engineering and Manager Electrical/Instrumentation and Control Engineering are combined into that of Manager Design and Analysis.

The positions of Senior Technical Advisor to VP of Nuclear Engineering Programs and Project Control, Advisor to the Director Nuclear Engineering, Project Management and the Secretaries for the Director Nuclear Engineering and VP Engineering and Project Controls have been removed.

These organization changes are consistent with existing management oversight of engineering activities performed for JAF.

JAF-SE-00-028, REV. 0: **Acceptability of Using Alternate Feedwater Temperature Inputs to the Reactor Heat Balance Calculation**

ENGINEERING CHANGE: **N/A**

This safety evaluation determined the acceptability of using alternate feedwater temperature inputs to the Reactor Heat Balance Calculation due to failure of one of the four Final Feedwater temperature elements that are normally used. RTDs are used to achieve a high degree of accuracy and reliability in measuring the temperature. Data from the RTDs was evaluated and compared to alternate methods and found to be very reliable and accurate.

To compensate for the failed feedwater temperature sensor this evaluation established the acceptability of either:

1) Using an operator input substitute value that is based on the remaining operable feedwater temperature indications.

Or

2) Provide an alternate input to the plant process computer from a known good source (i.e. another operable temperature element on the same Feedwater line).

This evaluation found that neither of these methods results in an unreviewed safety question or violation of the licensing basis. Both of these methods provide an acceptable means of compensating for a failed input to the automatic heat balance calculation.

JAF-SE-00-029, REV. 0: **Evaluation of the Railroad and Truck Access Port Concrete Roof Thickness**

ENGINEERING CHANGE: **N/A**

The purpose of this safety evaluation is to show that the update of the UFSAR with respect to the reinforced concrete roof thickness for the railroad and truck access port does not represent an unreviewed safety question. The revision to the UFSAR will indicate that the railroad and truck access port reinforced concrete roof varies in thickness from approximately 15 1/2 inches to 18 inches. The actual roof thickness for the railroad and truck access port as described in this evaluation does not adversely affect any structure, system or component (SSC) assumed to operate during normal plant operation or an accident. The actual roof thickness has been qualified in accordance with NYPA calculation JAF-CALC-BYM-04095. There are no physical or functional changes in any structure, system, or component, or changes to the design criteria. Changes evaluated by this NSE are consistent with current regulatory guidance. The evaluation demonstrates that the activity does not constitute any unreviewed safety question as defined by 10 CFR 50.59 nor does it involve significant hazards as defined in 10CFR50.92. Therefore, the activity can be performed as proposed.

JAF-SE-00-030, REV. 0: **Allowance of Temporary Setpoint Adjustment of Individual CRD Temperature Alarms**

ENGINEERING CHANGE: **N/A**

The control rod drives (CRD) at JAF are monitored for temperature. The alarm setpoint for each individual CRD is 250°F increasing. At times, depending on maintenance and wear on components as well as system configuration, individual CRDs may not be supplied adequate cooling water to have the individual CRD temperature remain below the 250°F alarm setpoint. Some CRDs will have more than adequate cooling flow while starving the other CRD. When this occurs operations follows a procedure to reestablish cooling water to the CRD exhibiting the high temperature. Occasionally this evolution fails to reduce temperature and the CRD that is in alarm stays in alarm. Though not the preferred way to operate, continued operation with the hot CRD is warranted but with concern for seal life at elevated temperatures.

When a CRD temperature alarm is sealed in, the other CRD temperature alarms are masked by the constantly alarming annunciator. Operations must periodically consult the temperature recorder to extract CRD high temperature alarm information. To allow automatic alarming of CRD high temperature conditions, a temporary setpoint change from 250°F to 300°F for the alarmed CRD may be implemented to bring the annunciator out of alarm.

GE SIL 173 gives guidance to determine the potential detrimental effects that operation with various elevated CRD temperatures might produce. The effects are primarily reduced life of the CRD seals and slightly increased scram times (less than 4% at 400°F). The proposed temporary setpoint of 300°F is well within the guidelines such that the high temperature effects to the seals will be minimal and there should be no effect on scram times. No effects are pointed out in the SIL at the proposed temperature limit (300°F) that is not bounded by the safety analysis as described in the JAF licensing basis. No effects are described in this evaluation that will impact any safety margins as defined in the JAF Technical Specifications.

In addition, allowance for the temporary setpoint is made for a maximum of 10 CRDs. The Improved Technical Specifications call out the limit for slow rods to be 10 maximum. A CRD that has a temperature above 350°F is to be considered a slow rod until it can be verified by testing that it is not slow. For CRDs that are over the normal high temperature of 250°F and are operating as designed and within the temporary setpoint of 300°F, a quantity of ten maximum was deemed a reasonable limit. A temporary temperature setpoint of 300°F for a maximum of 10 CRDs will have no detrimental effects on the operation of the over temperature CRD and as a result no effect on public safety.

The statement in AOP-69 that reads: "if 5 or more high temperature alarms exist at the same time within any 30 minute period then manually scram reactor and enter AOP-1". will be changed to read: "if 5 or more additional high temperature alarms (either 250°F or 300°F) come in and stay in within any 30 minute period then manually scram reactor and enter AOP-1".

JAF-SE-00-031, REV. 0: CRD Parts Equivalency

ENGINEERING CHANGE: JD-99-139

Design Change JD-99-139 "CRDM Parts Equivalency" provides the technical justification to use various replacement parts in Control Rod Drive Mechanisms (CRDMs) per the equipment vendor's recommendations. The CRDM replacement parts are of the same form, fit and function as the original parts. This design change also provided the technical justification to change the CRD cooling water differential pressure range while maintaining the same flow rate. As such, the CRDM replacement parts and reduction in cooling water differential pressure does not adversely impact any safety significant function. The changes proposed by this design change do not degrade the Security Plan, Quality Assurance Program, the Fire Protection Program, the Environmental Report or the Emergency Plan. This change does not constitute an Unreviewed Safety Question per 10 CFR 50.59.

JAF-SE-00-032, REV. 0: Extended Loadline Limit Analysis (ELLLA) Implementation

ENGINEERING CHANGE: N/A

This evaluation considered changes to the Core Operating Limits Report in support of operation under Extended Loadline Limit Analysis (ELLLA). Under this analysis operation to the APRM Rod Block line is permitted at any flow up to rated power, and the Flow-Biased APRM Rod Block and Scram lines are re-defined with a lower slope and higher bias consistent with the ELLLA and power uprate safety analyses. This expansion of allowed regions of operation on the power-flow map replaces limits established under the Loadline Limit analysis.

The effect of the wider flow range on transient and accident initial conditions is considered in the analysis and evaluation, and it is determined that existing fuel thermal limits provide adequate protection against unacceptable consequences of any accidents or transients affected by this change.

JAF-SE-00-034, REV. 0: **Acoustic Monitoring System "hard line" mineral insulated cable**

ENGINEERING CHANGE: **N/A**

The purpose of the safety evaluation is to accept the current installation of the "hard line" mineral insulated cable, which runs exposed for a short distance between the end of the conduit in which it is routed and the connection it makes to the acoustic (accelerometer) monitor on each main steam safety relief valve (MSSRV) "tail pipe". This condition is non-conforming to the UFSAR section 7.1.9.a, as the "hard line" cable is not routed entirely in conduit all the way to the accelerometer.

The conduit for each "hard line" cable is run as close as practical to the "acoustic" vibration accelerometer. The "hard line" cable then leaves the conduit for a short distance and terminates on the accelerometer. This "hard line" mineral insulated cable has a solid stainless steel jacket (shield), which surrounds a central conductor with magnesium oxide insulating material. The conduit is installed for this cable to provide a means to protect the cable from physical damage and to support the run of this cable from the accelerometer to the junction box. This "hard line" mineral insulated cable is capable of sustaining the environmental conditions within the containment without the need for the protection that is afforded by the conduit in which it is run. The "hard line" mineral insulated cable is connected to the accelerometer using an environmentally qualified connection. There are no deviations from the Cable Separation Criteria associated with the installation of the "hard line" cables, as all the "hard line" cables are non-safety related.

An unreviewed safety question is not involved because the "hard line" cables in free air are environmentally qualified and the use of conduit for the subject installation is not required to assure functionality of the safety relief valve position monitoring circuits. Additionally, no requirements exist in the JAF electrical separation criteria, 10CFR50 Appendix R or in the accident analysis that requires the use of metallic conduit for the subject installation.

JAF-SE-00-035, REV. 0: **Justification for Continued Operation of the Service Air System per NRC IE Bulletin No. 80-10**

ENGINEERING CHANGE: **N/A**

This Nuclear Safety Evaluation is being done to ensure that continued operation of the Service Air system when radiologically contaminated does not involve an unreviewed safety question or change to Technical Specifications. NRC 1E Bulletin No. 80-10 requires that the 10 CFR 50.59 evaluation consider the level of contamination (concentration and total curie content) and any potential releases of radioactivity to the environment. Potential releases shall be compared to 10CFR20, the facilities technical specifications and the environmental dose limits of 40CFR190. Potential release points are required to be monitored and controlled and maintained to as low as reasonably achievable. Based on the fact that the Service Air system is able to function as intended and systems that could potentially be cross-contaminated from the Service Air system will continue to perform their normal function, no unresolved safety question exists.

JAF-SE-00-036, REV. 0: Service Water Hot Taps

ENGINEERING CHANGE: JD-00-076

Design Change JD-00-076 "Service Water Hot Taps" provides the technical justification to install additional drains in the Normal Service Water System by the hot tap method. The Normal Service Water System is classified as QA Category II/III. As such, the Normal Service Water System does not support any safety-related equipment or is required for safe shutdown. The changes proposed by this design change do not degrade the Security Plan, Quality Assurance Program, the Fire Protection Program, the Environmental Report or the Emergency Plan. This change does not constitute an Unreviewed Safety Question per 10 CFR 50.59.

JAF-SE-00-038, REV. 0: Perform Weld Overlay Installation

ENGINEERING CHANGE: N/A

The purpose of this safely evaluation is to show that the installation of weld overlays to repair welds that contain IGSCC type cracks does not represent an unreviewed safety question. The evaluation shows that the weld overlay design and installation is a proven process and has been installed on primary pressure boundary piping since the early 1980's with no adverse operational consequences. Inspection results show that there has been no crack growth into weld overlays due to IGSCC. The installation of the weld overlay on welds that contain IGSCC type defects restores original design margins as required by ASME XI and applicable Code Cases. The weld metal is resistant to IGSCC crack growth and the installation process results in *a* compressive residual stress pattern, which also retard future IGSCC crack growth. The installation of weld overlays is allowed by the ASME Code via Code Cases and has been approved for use by the NRC on a plant specific basis.

The installation of the weld overlay does not adversely affect any structure, system or component (SSC) assumed to operate during normal plant operation or an accident. There are no functional changes in any structure, system, or component, or changes to the design criteria, The physical change to install a weld overlay to repair welds that contain IGSCC is a proven method and restores design margin. Changes evaluated by this NSE are consistent with original design requirements and are allowed by the ASME Codes. The evaluation demonstrates that the activity does not constitute any unreviewed safety question as defined by 10 CFR 50.59 nor does it involve significant hazards as defined in 10 CFR 50.92. Therefore, the activity can be performed as proposed.

JAF-SE-00-039, REV. 0: 12MOV-18 Actuator Closing Logic

ENGINEERING CHANGE: JD-00-078

The Reactor Water Cleanup (RWCU) Outboard Isolation Valve has a history of Appendix J local leak rate failures. DER 97-00528 documents the failure history of the valve. It was determined based on industry experience and the vendor recommendation that the valve closing logic should be changed from torque control to limit switch control. Design change JD-00-078 provides the technical justification for this alternative.

Design Change JD-00-078, will modify the closing logic for 12MOV-18(OP). The logic change will place a spare position contact in series with the closing coil. The existing 2 and 2 rotor will be replaced with a 3 and 1 rotor. The limit switch will be adjusted to open and stop the MOV when the closed seat is contacted. The limit switch adjustment will prevent the additional torque from being applied when closure is based on torque switch setting (existing configuration). The additional torque is causing the seating surfaces to deform and leak. This action will allow the valve to be closed to a precise point based on actual stem position rather than on torque.

This safety evaluation has shown that the installation and testing of design change JD-00-078 does not represent an unreviewed safety question.

JAF-SE-00-040, REV. 1: **Temporary Modification for Additional Air Compressor – Electric**

ENGINEERING CHANGE: **N/A**

Temporary Modification TM-00-060 installs a temporary electric motor driven oil-free air compressor and air dryer outside of the turbine building with hose routed to plant air systems upstream of the instrument air dryers (at valve 39AS-2D). This temporary modification will provide an additional source of high quality air to support maintenance and modification activities on the normal plant air compressors.

JAF-SE-00-041, REV. 0: **Change of AP-01.04 Action Statements When Multiple Crescent Area Unit Coolers Are Inoperable**

ENGINEERING CHANGE: **N/A**

Changing the shutdown action statement in AP-01.04 "TECH SPEC RELATED REQUIREMENTS, LISTS AND TABLES" for more than one crescent area unit cooler inoperable from 24 hours to 7 days does not involve an unreviewed safety question.

The crescent area coolers provide a cooling function for the environment where safety related equipment must operate after a design basis accident. Since reactor operation is permissible, by original licensing and technical specifications for up to seven days with no ESW cooling available to the crescent area coolers in one crescent, operation for up to 7 days with all crescent area coolers inoperable in one crescent is also acceptable.

JAF-SE-00-042, REV. 0: Ninety Day Grace Period for Internal QA Audits

ENGINEERING CHANGE: N/A

The changes proposed by this Nuclear Safety Evaluation involve adding a note to the end of Section 17.2, Appendix E, Paragraph 17.2E.2 of the JAFNPP FSAR allowing a 90 day grace period for completion of internal QA audits.

The changes are administrative in nature and do not involve plant equipment or plant operating conditions. The changes do not reduce the effectiveness of the QA Program, management of activities, or the oversight of plant operations. Therefore, these changes do not involve an unreviewed safety question.

JAF-SE-00-043, REV. 0: **Long Term Operation in Single Element Feedwater Level Control**

ENGINEERING CHANGE: **N/A**

The purpose of this safety evaluation is to determine whether long term operation of the Feedwater Control System in single-element control, when the plant is above 25 percent power, constitutes an unreviewed safety question.

Current plant design and licensing basis (FSAR Sections 1.6.4.2, 7.10.3, and 7.10.3.4.b) is inconsistent with Feedwater Control System operation in single element control at power levels > 25%. The proposed activity will clarify the plant design and licensing basis to ensure consistency with the FSAR in allowing unrestricted operation of the Feedwater Control System in single element control at power levels above 25%.

The update of FSAR sections 1.6, 7.10 and 10.8 will provide flexibility of operation in either single element or three-element control and to enhance the description of operation of the Feedwater Control System. This activity will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the SAR, will not create the possibility of an accident or malfunction of equipment important to safety of a different type as previously evaluated in the SAR, and will not reduce the margin of safety as defined in the basis of any Technical Specification. The design function and operation of the Feedwater Control System are not changed. Operation in the single element control will not cause a loss of feedwater or feedwater controller malfunction transient. Although level oscillations have been observed, existing transient analysis bound operation in single element control for power >25%. Based on discussions with General Electric (Telcon between NYPA and GE, dated 8/8/2000) there would be no significant difference between single or three element control on plant response for the Abnormal Operational Transients (AOTs) associated with the Feedwater Control System described in the JAF FSAR and the Transient Analysis Design Report (GEZ-6408 JAFNPP Transient Analysis Design Report, dated April 1975) The transient response relative to Feedwater Control System will not significantly affect scram avoidance. The JAFNPP Transient Analysis (NEDM-11071-54 dated January 1974, J11-03359SRL Rev. 1 dated October 1998, and GE-NE-187-58-1191 dated November 1991) evaluated the plant response to various Feedwater System transients. These reports demonstrate plant capability to mitigate the consequences of these transients regardless of Feedwater Control System mode of operation.

JAF-SE-00-044, REV. 0: TIP Prox. Switch Program Modification

ENGINEERING CHANGE: N/A

The TIP system central control unit operating program will be modified to allow the system to operate without proximity switches. Additionally the failed proximity switch wiring to the motor control unit will be disconnected. The impact of this modification will be to change the electrical interlock, which requires the TIP probe to be withdrawn to below the indexer prior to rotating the indexer to the next channel for testing, such that this function is now performed by the software based on the encoder position. Secondly, this removes the ability of the failed proximity switch indication from stopping the insertion of the TIP probe into a given channel.

None of the above actions result in an unreviewed safety question.

JAF-SE-00-045, REV. 0: Cycle 15 Reload Core

ENGINEERING CHANGE: JD-99-091

This evaluation documents the changes made to the reactor core by the refueling prior to the start of Cycle 15. The safety analyses performed on the Cycle 15 core have been reviewed and found adequate in scope and content against the requirements in GE Report, General Electric Standard Application for Reactor Fuel (GESTAR) (NEDE-24011-P-A-14, June 2000) and the FitzPatrick SAR and Technical Specifications. Fuel design changes have been reviewed against designs previously used at FitzPatrick. On the basis of this review no unreviewed safety question exists, and the safety analyses properly support the limits established in the Core Operating Limits Report (COLR). Based on the foregoing summary it can be concluded that operation of FitzPatrick in the Cycle 15 core as given in the reference documents, to the limits given in the COLR does not constitute an unreviewed safety question.

The five replacement control rod blades (CRB) installed under this modification have been evaluated against the SAR. There are no Technical Specification requirements affected by this modification. Design differences between these CRBs, other replacement CRBs and the original equipment were found to be insignificant to their interface with other plant components and their function. Based on these conclusions the replacement CRBs are acceptable.

JAF-SE-00-046, REV. 0: Control Rod Blade Replacement (with fuel in core)

ENGINEERING CHANGE: N/A

The purpose of this safety evaluation is to determine whether the combined grapple including the quiver if used during refueling outages with fuel in the reactor core, constitutes an unreviewed safety question.

The proposed activity will clarify the plant design and licensing basis to ensure consistency with the FSAR in allowing unrestricted use of the combined grapple/quiver during refueling outages.

The update of FSAR section 9.17.5 will provide flexibility of In-vessel servicing equipment during refueling outages. This NSE addressed the use of the combined grapple/quiver and determined that under the worse condition or malfunction of this equipment the event would be bounded by the FSAR Refueling Accident. The Refueling Accident consequences and plant capability to mitigate the consequences of this event regardless of this in-vessel equipment usage during refueling periods are unchanged.

JAF-SE-00-047, REV. 0: **Use of Trailer Mounted Water Treatment Skid to Supply Demineralized Water to the Plant Demin. Water Storage Tanks**

ENGINEERING CHANGE: **N/A**

This activity is proposed to enhance the availability of Demin Makeup Water Treatment System during refueling outages when the Decay Heat Removal system becomes a large load. The activity involves the use of a trailer mounted water treatment skid to perform the de-ionization process to maintain full the Demin Storage Tanks located in the screenwell. It is concluded by this analysis that this activity does not constitute an unreviewed safety question because the worst case scenario is a complete loss of demin makeup water supply. Since this function is not important to safety, and no SSC's are involved that are important to safety, there is no unreviewed safety question.

JAF-SE-00-048, REV. 0: RO14 Temporary Power to 17RM-452A

ENGINEERING CHANGE: N/A

This activity involves a temporary plant modification to disconnect 17RM-452A from its normal 120VAC power supply, the 'A' RPS bus. Temporary power will then be supplied to 17RM-452A through connection to 71LPEC1 via a 120VAC utility receptacle.

This safety evaluation considers the use of the alternate power supply in determining that no unreviewed safety questions exists. The alternate power supply is directly fed from the same source as the power supply to the 'A' RPS bus and meets the power specification requirements of the radiation monitor to support its design and licensing functions. The source of the power supply to the radiation monitor was determined to have no effect on the ability of the radiation monitor to perform its intended function and to have no effect on how the radiation monitor performs its function. Additionally, disconnecting the radiation monitor from the 'A' RPS bus prior to connecting the alternate power supply to the radiation monitor was evaluated to ensure that this change did not impact any other components supplied by the 'A' RPS bus. 17RM-452A will be considered operable in this temporary configuration.

The source of the power supply was determined to have no impact on the function of the radiation monitor. As such, the evaluation of this change supports that the radiation monitor will function identically to the manner in which it functions in the configuration shown on FSAR Figure 7.12-1 and that this change has not resulted in an unreviewed safety question.

JAF-SE-00-049, REV. 0: **Evaluation of Identified Indication in the "B" Core Spray Piping Inside the Reactor Pressure Vessel**

ENGINEERING CHANGE: **N/A**

During R014, an indication was identified during In-Vessel Visual Inspection of the Core Spray piping internal to the Reactor Pressure Vessel. An engineering analysis of the indication evaluated the structural adequacy of the 'B' Core Spray piping internal to the RPV considering conservative as-found dimensions and bounding growth in size over one operating cycle. The analysis also determined the bounding leakage rate using differential pressure across the crack based on Core Spray flow rate of 6850 gpm. Operation with the indication, as evaluated, has been determined to be within the current licensing basis and is therefore acceptable. The condition will not be repaired at this time. Future In-Vessel Visual Inspections will be used to confirm the analysis conclusions remain valid.

JAF-SE-00-051, REV. 0: **Operation of the Drywell Continuous Atmosphere Monitor (DW CAM) System with Flow Less than 10 CFM**

ENGINEERING CHANGE: **N/A**

The UFSAR states that the DW CAMS flowrate is 10CFM. This value was in error since the original FSAR. Furthermore, due to ongoing system flow instabilities in the Drywell Atmosphere Radioactivity Monitor System (DW CAMS), one of the two particulate monitoring pumps in each train was de-energized. This corrected the condition of flow instabilities and reduced the total flowrate to the DW CAMS from approximately 6 to approximately 4 CFM. This is because the normal flowrate from the particulate monitor was reduced from 4 CFM to between 2 and 3 CFM. The Drywell Atmosphere Radioactivity Monitor System (DW CAMS) exists to monitor the airborne radioactivity in the drywell atmosphere. It consists of particulate, iodine, and gaseous monitors. A calculation to determine reasonable flow limits for the DW CAMS concluded that the system is not capable of attaining the UFSAR flow of 10 CFM but will work optimally with a flow rate through the particulate subsystem of 1.0 to 5.0CFM. The calculation also clarified the capabilities of the gaseous monitor relative to the ability to detect leakage of reactor coolant from a steam leak only.

This evaluation concludes there are no unreviewed safety questions involved with operating the DW CAMs with a flowrate of 1.0 to 5.0CFM in the particulate subsystem. The noble gaseous monitoring side of the DW CAMs is insensitive to this change in flowrate to the particulate subsystem and is only capable of detecting noble gases from a steam leak. The Particulate subsystem is linearly dependent on flowrate and has the required sensitivity to be able to detect a leak below the established leakage rate limits. The time to detect this leakage is primarily dependent on the reactor coolant activity level, the size of the leak and the location of the leak.

JAF-SE-00-052, REV. 0: **Temporary Modification for Radwaste Strainer Installation (TM-00-070)**

ENGINEERING CHANGE: **N/A**

The installation of one 150 lb. Class, flanged "y"-type strainer within the Radwaste system discharge line serves to enhance the capability of the Radwaste system to discharge fluid to the lake that is within specified environmental limits. This Temporary Modification (TM- 00-070) does not adversely affect any equipment important to safety and is considered QA Category II/III, non-safety related, and non-seismic . The overall design and method of operation for the liquid Radwaste system has not been altered other than the added capability to remove potentially existing resin beads from the Radwaste discharge flow path that would normally travel to the lake.

As no resin beads have been discovered in the "A" and "B" tanks, this temporary strainer is not expected to remove any appreciable amount of resin beads. Rather, it is intended to add an additional conservative capability for maintaining controlled and clean discharge to the lake. The capability to "blow-down" the "y"-type strainer gives an additional measure of control over any potential resin beads that may remain in the affected piping system, as the strainer contents can be forced out of the strainer collection area via pressure from the radwaste discharge pump(s). This will allow the flexibility of continual monitoring of strainer contents if so desired.

JAF-SE-00-053, REV. 0: Entergy Nuclear Northeast Day-One Organization

ENGINEERING CHANGE: N/A

The purpose of this NSE is to describe and evaluate proposed changes to the organizational structure, as described in the IP3 and FitzPatrick UFSARs. These changes are the result of the transfer of the IP3 and FitzPatrick operating licenses from the New York Power Authority to Entergy Nuclear Operations, Inc.

Specifically, these changes include:
- Adoption of a new, top-tier corporate organization primarily affecting the headquarters office.
- Change the title Site Executive Officers to Vice President Operations -- The Vice President Operations – JAF and Vice President Operations - IP3 will fulfill the roles and responsibilities of the IP3 and FitzPatrick Site Executive Officer (SEO). Minimum qualifications for both positions will be the same.
- Change the title Director Regulatory Affairs and Special Projects (DRA&SP) to Director Licensing. The Director Licensing will fulfill the roles and responsibilities of the DRA&SP.
- Creation of several corporate support positions including: I Vice President Operations Support
- Vice President Communications & Government Affairs

These changes do not eliminate any functional requirements. The proposed changes are administrative in nature and do not involve plant equipment or operating conditions. They will not reduce the effectiveness of the management of activities or of the oversight of plant operations. Therefore, these changes do not involve an unreviewed safety question.

JAF-SE-00-054, REV. 0: **Revised Trip Level Setting for 14DPIS-43A and 14DPIS-43B, Above Core Plate to Core Spray Line at RPV dP**

ENGINEERING CHANGE: **N/A**

This evaluation considered whether it is acceptable to change the Trip Level Setting required by AP-01.04 for differential pressure indicating switches which monitor the integrity of the core spray lines between the RPV wall and the core shroud from the present ≤ 0.5 psid to ≤ -0.2 psid. A calculation (JAF-CALC-CSP-04231, "Equivalent Analytical Limit for 14DPIS-43A and 14DPIS-43B," Revision 0) has determined that the trip setting must be ≤ -0.2 psid to ensure the function of the switch. The evaluation also supports editorial changes which more accurately reflect the design of the instruments. The changes have been determined to be within the current licensing basis and are therefore acceptable.