Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
James A. Fitzpatrick NPP
P.O. Box 110
Lycoming, NY 13093
Tel 315 349 6024 Fax 315 349 6480

T.A. Sullivan
Vice President, Operations-JAF

May 12, 2003
JAFP-03-0065

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station O-P1-17
Washington, DC 20555

SUBJECT: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Summary of Plant Changes, Tests, and Experiments for 2001/ 2002 as Required by 10 CFR 50.59 and 10 CFR 72.48, and Summary of Commitment Changes for 2001/2002

Dear Sir or Madam,

This letter transmits a summary of changes, tests, and experiments implemented at the James A. FitzPatrick Nuclear Power Plant for the years 2001 and 2002 as required by 10 CFR 50.59 (d) (2) and 10 CFR 72.48 (d) (2). Also included is a summary of revised regulatory commitments as require by Nuclear Energy Institute Guideline NEI 99-04, Guidelines For Managing NRC Commitment Changes, endorsed by the Commission in NRC Regulatory Issue Summary 2000-17.

Attachment I provides a summary of each 10 CFR 50.59 report including a brief description of the changes, tests, and experiments, including a summary of the evaluation of each. Also included is the 10 CFR 50.59 evaluation number (e.g. JAF-SE-01-001), revision number (if applicable), title, activity type, and engineering change number (if applicable).

Attachment II provides 10 CFR 72.48 reports containing a brief description of the changes, tests, and experiments, including a summary of the evaluation of each.

Attachment III provides summaries of regulatory commitment changes requiring NRC notification, and a brief statement of the basis for the change. Also included is the FitzPatrick tracking number of the change, and the basis document from which the commitment was made.

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Should you have any questions concerning this report, please direct them to Mr. Andrew Halliday, Regulatory Compliance Manager, at (315) 349-6055.

Very truly yours,

for

T.A. Sullivan

TAS:GJB

cc: Regional Administrator
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Office of the Resident Inspector
U.S. Nuclear Regulatory Commission
James A. FitzPatrick Nuclear Power Plant
P.O. Box 136
Lycoming, NY 13093

Mr. Guy Vissing, Project Manager
Project Directorate I
Division of Licensing Project Management
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attachment I

Summary of Plant Changes, Tests, and Experiments for 2001/2002 as Required by 10 CFR 50.59

Entergy Nuclear Operations, Inc
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

ATTACHMENT I

Summary of Plant Changes, Tests, and Experiments for 2001/2002 as Required by 10 CFR 50.59

A. Introduction

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

> ...A licensee may make changes in the facility as described in the final safety analysis report (as updated), make changes in the procedures as described in the final safety analysis report (as updated), and conduct tests or experiments not described in the final safety analysis report (as updated) without obtaining a license amendment....

if the change meets the requirements as outline in sections 50.59 (c) (1).

10 CFR 50.59 (d) (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 evaluation of each.

Unless otherwise noted, each 10 CFR 50.59 evaluation listed concluded that it's subject change, test, or experiment did not:

- Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the final safety analysis report;
- Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the final safety analysis report;
- Result in more than a minimal increase in the consequences of an accident previously evaluated in the final safety analysis report;
- Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the final safety analysis report;
- Create a possibility for an accident of a different type than any previously evaluated in the final safety analysis report;
- Create a possibility of a malfunction of an SSC important to safety with a different result than previously evaluated in the final safety analysis report;
- Result in a design basis limit for a fission product barrier as described in the final safety analysis being exceeded or altered; or
- Result in a departure from a method of evaluation described in the final safety analysis report used in establishing the design bases or in the safety analyses;

and therefore, did not require prior NRC approval prior to implementation.

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Summary of Plant Changes, Tests, and Experiments for 2001/2002 as Required by 10 CFR 50.59

JAF-SE-94-022, REV. 1: **Seismic Verification of Equipment by SQUG Generic Implementation Procedure**

Activity Type: **N/A**

The purpose of this evaluation is to confirm that the use of the Generic Implementation Procedure (GIP-2) for Seismic Verification of Nuclear Plant Equipment prepared by SQUG for the seismic design and verification of existing, modified, new and replacement equipment does not constitute an unreviewed safety question. Based on this evaluation, the James A. FitzPatrick FSAR will be revised to include this method as an alternate approach for equipment seismic verification.

The nuclear safety evaluation determined that no unreviewed safety question is created by the FSAR change and therefore the use of the GIP methodology is acceptable. The use of the GIP will not affect the ability of safety-related equipment or equipment important to safety to perform required safety functions during or after a seismic event.

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Summary of Plant Changes, Tests, and Experiments for 2001/2002 as Required by 10 CFR 50.59

JAF-SE-97-039, REV. 2: **Torus/Drywell Vacuum Breaker Alternate Test Method and Review of Primary Containment Inerting and Deinerting Operations**

Activity Type: **N/A**

Periodic testing of the pressure suppression chamber (torus)/drywell vacuum breakers requires equalizing the (intentional) differential pressure between the two containment air spaces. This evaluation supports a change to the method of accomplishing this test precondition. The existing UFSAR test description permits the pressure suppression function to be bypassed prior to performing the required testing. The LOCA type of accident relies on the proper performance of containment air space energy paths to ensure acceptable accident response. The proposed test method will improve containment loading in the event of a LOCA over the current test methodology while the equalization is in progress. The UFSAR description for section 5.2.3.6 is being revised.

The additional restriction on test methods will not increase the probability of a LOCA and will maintain LOCA consequences within the existing analyses by ensuring expected containment response. The systems selected to provide the air space pressure equalization have been designed for the functions they will be serving. No new malfunction exists. The systems selected to equalize the pressure are performing existing, evaluated roles that will not create any new accident or malfunction.

Revision 2 to the evaluation makes a correction to the UFSAR regarding inerting and deinerting both the drywell and suppression pool exhaust and supply lines and not merely the suppression chamber supply line and the drywell exhaust line. Revising the UFSAR text to make certain that a suppression function bypass condition during inerting and deinerting operation will not exist ensures consistency with the LOCA analyses and the plant design basis documents. This revision also treated "consequences" as "dose" instead of a "penalty". The restoration of design LOCA assumptions will ensure no reductions in the margin of safety are created.

Based on the above conclusions, NRC review is not required.

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JAF-SE-99-002, REV. 4: **Reactor Building Crane Upgrade**

Activity Type: **Design 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 are 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 10CFR50.59 nor does it involve significant hazards defined in 10 CFR 50.92. Therefore, the activity can be performed as proposed.

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JAF-SE-99-029, REV. 0: **Durability Monitor System**

Activity Type: **Design Change JD-99-142**

Design Change JD-99-142 installs the Durability Monitor System (DMS) in the Reactor Building and addresses the removal of the Crack Arrest Verification (CAV) System Load Frame to make room for the DMS. Noble metals have been applied to the wetted surfaces of the reactor internals and connected piping. The Durability Monitor System (DMS) is one way to monitor the amount of noble metal remaining on these surfaces thereby providing data used to determine when NobleChem ™ should be reapplied. By demonstrating adequate noble metal deposition, redeposition or use of industry data and knowing the requisite amount of excess hydrogen in the reactor coolant, one can infer adequate protection from Intergranular Stress Corrosion Cracking (IGSCC).

Installation and operation of the DMS does not involve an unreviewed safety question nor does it require a change to JAF Technical Specifications.

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JAF-SE-00-003, REV. 0: **Update to FSAR to Remove Inconsistency Concerning Maximum EDG Room Temperature**

Activity Type: **FSAR Change**

The proposed changes to FSAR sections 7.1.12 and 9.9.3.1 and Table 9.9-1 update the design condition maximum temperature for Emergency Diesel Generator rooms based on test results and reanalysis. The proposed changes also correct misleading discussion concerning which areas are cooled by ventilation systems (as opposed to cooling water systems) and how redundancy is provided by the Emergency Diesel Generator Room Ventilation system.

Summary of Plant Changes, Tests, and Experiments for 2001/2002 as Required by 10 CFR 50.59

JAF-SE-00-008, REV. 1 — **Construction of Independent Spent Fuel Storage Installation**

Activity Type: — **Design 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 the 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 cast assembly pad, the roadway between the ISFSI storage pad and the cast 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.

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JAF-SE-00-011, REV. 0: **Power Supplies - 15MOV-102(OP)/103(OP)**

Activity Type: **Design Change JD-99-015**

Design Change Package JD-99-015 resolves safety train separation deficiencies and re-enables Reactor Building Closed Loop Cooling (RBCLC) System motor-operated valves 15MOV-102 and 15MOV-103, which isolate the Emergency Service Water (ESW) supply to the "A" and "B" Drywell Coolers, respectively. These normally closed valves were designed for manual operation from the Control Room or local operator panels to provide ESW to the Drywell Coolers if RBCLC is not available. The breakers associated with these two valves were opened in 1992 to prevent spurious operation due to a fire in various plant areas since there was no analysis or test data available at the time to confirm that adequate cooling would remain for safe shutdown loads if ESW flow were diverted to the Drywell Coolers. The recent development of a hydraulic network model for ESW demonstrates that a balanced system could support safe shutdown loads with flow diverted to the Drywell coolers via 15MOV-102 or 15MOV-103.

Power to 15MOV-102 and 15MOV-103 currently is supplied from the bus opposite the ESW train for which isolation is provided by each valve. The design change eliminates the current safety train separation deficiencies by powering each valve from the divisional power source associated with the ESW train isolated by that valve. The hydraulic analysis provides the basis for closing the breakers for both valves.

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JAF-SE-00-014, REV. 3:	**SRV Electric Lift/ATWS Setpoint Change Project**
Activity Type:	**Design 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; and (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.

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JAF-SE-00-025, REV. 4: **Provide Chemical Cleaning Process For ESW Piping and Heat Exchangers**

Activity Type: **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. The degradation of the flow rates has been worsening with time. The heat exchanger heat removal capability operability margins have been reduced. 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 plant procedures. 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. 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.

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JAF-SE-00-033, REV. 0:	**Revise AP-01.04 to Include Additional Early Warning Fire Detection Systems and the HPCI Foam System**
Activity Type:	**Procedure Change**

This nuclear safety evaluation justifies the addition of several early warning fire detection systems and the HPCI foam system to Administrative Procedure AP-01.04, " Tech Spec Related Requirements, Lists, and Tables", and clarifies the differences between early warning detectors and those that actuate suppression systems.

The additional early warning fire detection and HPCI foam systems are part of the NRC approved fire protection program as documented in several NRC Safety Evaluation Reports. These systems were not part of the original Technical Specifications that were transcribed verbatim into the initial issue of AP-01.04. The addition of the fire detection and HPCI foam systems to AP-01.04, will result in specific technical and surveillance requirements being applied to the systems. Compensatory measures will be defined in the event of system unavailability and future proposed changes to these systems will be reviewed within the bounds of the approved fire protection program. These program enhancements continue to demonstrate that appropriate fire protection features are in-place to satisfy the license condition acceptance criteria approved by the NRC.

This evaluation demonstrates that the addition of these systems to the approved fire protection program does not adversely affect the ability of the plant to achieve and maintain safe shutdown in the event of a fire.

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JAF-SE-00-037, REV. 0: **Evaluation of the Main Stack Location**

Activity Type: **N/A**

The purpose of this evaluation is to show that the update of the UFSAR with respect to the location and relationship between the Main Stack and other Class I (safety-related) structures does not represent any unreviewed safety question. The revision to the UFSAR will show that the location of Class I (safety-related) structures within the overall shadow of the Main Stack is acceptable based on the evaluated failure mechanism of the stack.

The location and relationship between the Main Stack and other Class I (safety-related) structures does not adversely affect any structure, system or component (SSC) assumed to operate during normal plant operation or an accident. The location and relationship between the Main Stack and other safety-related structures has been evaluated in accordance with calculation JAF-CALC-BYM-04122 and found to be acceptable.

There are no physical or functional changes in any structure, system, or components. Changes evaluated by this evaluation 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.

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JAF-SE-00-050, REV. 0: **Digital Upgrade – Replacement of DW Equipment and Floor Drain Sump Level Recorders 20LR-122A/ B**

Activity Type: **Design Change JD-00-075**

This evaluation determines the acceptability of replacing Drywell Equipment and Floor Drain Sump Level Recorders 20LR-122A & B with digital programmable recorders. The criteria of 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 were used as guidance. The evaluation considered the differences in technology between the original analog chart recorders and the digital replacement. Failure modes and effects of the new recorder including common cause software failures, common mode failures and effects of the firmware based components, effects of the Human-Machine Interface, and Electromagnetic Compatibility were analyzed in determining the acceptability of the replacement recorders. The evaluation also considered the adequacy of software controls and procedures. The recorders initiate no automatic actions, are not required to meet Technical Specifications, are not involved in mitigating or monitoring the consequences of any accident, and do not interact with any equipment important to safety. They provide trend data only. The evaluation concluded that the change does not create an unreviewed safety question.

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JAF-SE-01-001, REV. 1:	**Revision to AP-01.04 to Add LCO & Surveillance Requirements for SRV Electric Lift Instrumentation & Delete the Requirements for Technical Services Review**
Activity Type:	**Procedure Change**

The proposed activity is to revise Administrative Procedure AP-01.04, "Tech Spec Related Requirements, Lists, and Tables" to add LCO & surveillance requirements for SRV Electric Lift instrumentation and delete the requirements for Technical Services review.

The original revision of this safety evaluation established AP-01.04 LCO and surveillance requirements for the SRV Electric Lift System (SRVELS) and evaluated additional administrative changes to AP-01.04.

Revision 1 updates the evaluation to reflect inclusion of the SRV Electric Lift System in the UFSAR. Revision 1 also removes reference to memorandum JLIC-00-0057 from the evaluation and attached Limiting Conditions for Operation Bases providing more current evaluation description and Bases consistent with FitzPatrick Nuclear Safety Evaluation JAF-SE-00-014, SRV Electric Lift/ATWS Setpoint Change Project, Rev. 3 and the UFSAR.

Revision 1 further revises the evaluation to reflect the requirements and formatting of the current revision of MCM-4.2.

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JAF-SE-01-002, REV. 0: **Entergy Organizational Changes**

Activity Type: **N/A**

The organizational changes proposed by this Nuclear Safety Evaluation involve the creation of the position titles of General Manager Plant Operation, Director Safety Assurance, and Business Services Manager who will report to the VP Operations. The creation of these position titles is in parallel with the elimination of the position titles of Plant Manager, General Manager Support Services, General Manager Maintenance and the Financial Administration Manager. Additional changes are in support of JAF organizational restructuring following the plant license transfer to Entergy Nuclear Northeast. 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 operation. Therefore, the changes do not involve an unreviewed safety question.

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JAF-SE-01-003, REV. 0:	**Hydrogen/Oxygen Analyzer and Sensor Replacements**
Activity Type:	**Design Change JD-99-102**

This evaluation determines the acceptability of replacing dissolved hydrogen/dissolved oxygen analyzers in sampling panels 95SP-7 and 95SP-8 with digital programmable dissolved hydrogen/dissolved oxygen analyzers. The criteria of 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 10CFR50.59 were used as guidance. The evaluation considered the differences in technology between the original analog analyzers and the digital replacement. Failure modes and effects of the new analyzers including common cause software failures, common mode failures and effects of the firmware based components, effects of the Human-Machine Interface, and Electromagnetic Compatibility were analyzed in determining the acceptability of the replacement dissolve hydrogen/dissolved oxygen analyzers. The evaluation also considered the adequacy of software controls and procedures. The dissolved hydrogen/dissolved oxygen analyzers initiate no automatic actions, are not required to meet Technical Specifications, are not involved in mitigating or monitoring the consequences of any accident, and do not interact with any equipment important to safety. They provide indication and trend data only. The evaluation concluded that the change does not create an unreviewed safety question.

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JAF-SE-01-004, REV. 0: **Evaluation of Control Rod Fast Withdrawal Velocity – Cycle 15**

Activity Type: **N/A**

During performance of surveillance test ST-20N, Control Rod Exercise/Timing/Stall Flow Test, at the beginning of Operating Cycle 15, a control rod had a withdrawal time faster than normal and close to the specified time of 28.8 seconds. This corresponds to the maximum previously evaluated withdrawal speed of 5 inches per second specified in UFSAR Section 3.5.6.1.

GE and JAF 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. 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 15 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.

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JAF-SE-01-005, REV. 0:	**MSIV Limit Switch Instrument Functional Test With Failed RPS Position Switch**
Activity Type:	**Procedure Change**

The purpose of this evaluation is to provide assurance that temporary surveillance test procedure TST-109, MSIV LIMIT SWTICH INSTRUMENT FUNCTIONAL TEST WITH FAILED RPS POSTIION SWTICH **, is within prescribed limits. Technical Specifications require quarterly testing of the position switches (a reactor trip function) however with a failed switch it is not possible to test the remaining switches without performing a temporary modification which is accomplished by the procedure.

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JAF-SE-01-006, REV. 0:	**Hydrogen and Oxygen Injection Flow Recorder Replacements**
Activity Type:	**Design Change JD-01-024**

This evaluation determines the acceptability of replacing analog Hydrogen and Oxygen Injection Flow Recorders 89A-FR-100 and 89A-FR-300 with digital programmable recorders. The criteria of 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 10CFR50.59 were used as guidance.

The evaluation considered the differences in technology between the original analog chart recorders and the digital replacement. Failure modes and effects of the new recorder including common cause software failures, common mode failures and effects of the firmware based components, effects of the Human-Machine Interface, and Electromagnetic Compatibility were analyzed in determining the acceptability of the replacement recorders. The evaluation also considered the adequacy of software controls and procedures. The recorders initiate no automatic actions, are not required to meet Technical Specifications, are not involved in mitigating or monitoring the consequences of any accident, and do not interact with any equipment important to safety. They provide indication and data recording only. The evaluation concluded that the change does not create an unreviewed safety question.

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JAF-SE-01-008, REV. 1:	**Revision to AP-01.04 to add the Offgas Treatment System Explosive Gas Monitoring Program and Decrease the Low Flow Setpoint for the Associated Dilution Steam Flow Instrumentation**
Activity Type:	**Procedure Change**

It is proposed that Administrative Procedure AP-01.04, Tech Spec Related Requirements, List and Tables, be revised to add the Offgas Treatment System Explosive Gas Monitoring Program including the addition of LCOs, SRs, LCO Action statements and associated Bases as extracted from Section 3.7 of RETS and authorized per License Amendment No. 270. Additionally, the offgas dilution steam flow instrumentation low flow trip setpoint, which would reside in AP-01.04 as a result of the first activity, is proposed to be changed from 6300 lb/hr to 4800 lb/hr.

The existing process design dilution steam flow of 6770 lb/hr with a low flow trip of 4800 lbs/hr (70% of design) will provide adequate operating margin to ensure against spurious system trips. While the low flow trip setpoint change does depart from the guidance provided in GE SIL 150, it does so only to the extent to credit the presence of dilution steam in determining the actual flammability of a hydrogen-air-steam mixture. GE acknowledged that the 4% flammability limit used for hydrogen in SIL 150 was based on a hydrogen-air mixture and provided separate guidance to ensure the intent of SIL 150 would be fulfilled in the proposed low flow setpoint. The requirement to maintain hydrogen concentration below 4% limit specified in SIL 150 is not altered for that portion of the offgas treatment system downstream of the recombiner. Continuous operation just above this trip setpoint results in operating temperatures with adequate margin below the allowable system design limit (946°F vs. 1000°F, a margin of 54°F) to prevent overheating or ignition of the catalyst. Also adequate margin is provided below the flammability limit for hydrogen in a steam-diluted environment (5.6% vs. 7.3%). Therefore, the proposed activity was shown not to affect the ability of the Offgas System to meet its safety design bases as described in FSAR Section 11.4.3 nor change the assumptions or conclusions contained in the plant safety analysis.

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JAF-SE-01-009, REV. 0: **Feedwater Heaters Maximum String Flow**

Activity Type: **FSAR Change**

FSAR Section 10.8.3, Condensate and Feedwater System Description, states in part "In addition, the system is designed so that 70 percent of rated flow can be attained with three condensate pumps and three condensate booster pumps operating with one string of feedwater heaters in service". This UFSAR statement will be changed to "In addition, the system is designed so that *55* percent of rated flow ($6X10^6$ lbm/hr) can be attained with two or three condensate pumps and two or three condensate booster pumps operating with one string of feedwater heaters in service".

The proposed activity only affects the secondary side of the plant, which is not required for the safe shutdown of the plant.

Feedwater heater flow has little impact on accidents, although two transients would benefit from the proposed change. A reactor vessel water temperature decrease (loss of feedwater heating transient) is the most affected transient. The feedwater flow is directly related to the magnitude of the power level. Changing feedwater flow from 70% to 55% will have a less severe impact on this transient since the magnitude of the power rise decreases with the initial power condition (FSAR Section 14.5.3.1). An event resulting in a reactor vessel coolant inventory decrease could be a loss of feedwater (flow), however this transient is most severe at high power operation. A loss of feedwater at 55% feedwater flow instead of 70% would not be as severe since the rate of level decrease is lower and the amount of stored and decay heat to be dissipated is also less (FSAR Section 14.5.5.3).

All SSC will remain unchanged, only the method of operation will change. The design function will not be degraded by this proposed activity. Since SSC will be operated within their design limits, no new malfunctions are created.

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JAF-SE-01-010, REV. 0: **Operation of Independent Spent Fuel Storage Installation (ISFSI)**

Activity Type: **Procedure Change**

The purpose of this evaluation is to determine whether operation of the JAF Independent Spent Fuel Storage Installation (ISFSI) will require an amendment to the JAFNPP operating license prior to implementation. Holtec International Inc. HI-STORM System casks will be used to store the spent nuclear fuel assemblies in the JAF ISFSI. The activities covered by this evaluation include the operating procedures that will be used to perform the following: control of heavy loads; transport and assembly of the cask system; placement of an empty fuel transfer cask (HI-TRAC) Into spent fuel pool; loading the cask with spent nuclear fuel assemblies; draining, drying and sealing the spent fuel canister; transfer of the canister from the HI-TRAC transfer cask to a HI-STORM storage cask; on-site transport of HI-STORM System casks; and storage of loaded HI-STORM casks on the JAF ISFSI storage pad. Cask unloading operations are also covered by this evaluation.

Dry cask storage operations will interface with JAFNPP structures, systems and components including the following: Spent Fuel Pool; Reactor Building; Reactor Building Crane; Refueling Platform Fuel Grapple Hoist; Liquid and Gaseous Radioactive Waste Systems; Electrical System; Service Air System; and Site Roadways and Railroad Spur. In addition, gaseous effluents from cask draining, drying and unloading operations may be discharged to the Standby Gas Treatment System.

Based on the results of this evaluation, the proposed activity does not meet any of the criteria in 10 CFR 50.59(c)(2) and a change to the technical specifications is not required. Therefore, implementation of the ISFSI does not require prior NRC approval.

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Summary of Plant Changes, Tests, and Experiments for 2001/2002 as Required by 10 CFR 50.59

JAF-SE-01-011, REV. 0: **High Range Effluent Monitors Replacement**

Activity Type: **Design Change JD-01-021**

Maintenance personnel at JAF have identified a series of failures with the existing Victoreen High Range Effluent Monitors (HREMs) located in the Turbine Building (17R-434A and 17R-434B), Radwaste Building (17R-463A and 17R-463B), and Main Stack (17R-53A and 17R-53B). Failures to these channels and their associated ratemeters are due to faulty reed switches on the low and high range circuit boards. This root cause has since been confirmed with the vendor. The vendor has further stated that reed switches and circuit boards meeting Victoreen and JAF requirements are no longer produced and the design is considered obsolete. This design change will replace existing Victoreen High Range Effluent Monitors (HREMs) located in the Turbine Building (17R-434A and 17R-434B), Radwaste Building (17R-463A and 17R-463B), and Stack (17R-53A and 17R-53B) and their associated ratemeters in Control Room Panel 09-2.

The HREM RMS channels modified by this change provide the same functions as the existing components. They do not perform any automatic functions, do not interface with any components which can cause an accident initiation or equipment malfunction, and do not perform any automatic functions which could affect any normal effluent release termination functions. Postulated channel failures due to the use of new digital components within these channels will not result in any different types of accidents or malfunctions than those already postulated. This change does not result in an increase in the consequences of an accident or malfunction as evaluated in the FSAR. This change does not result in a design basis limit for a fission product barrier being exceeded or altered. The change does not result in a departure from a method of evaluation described in the FSAR.

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JAF-SE-01-012, REV. 0:	**Clarification of FSAR Description of EDG Air Start System**
Activity Type:	**FSAR Change**

This evaluation reviews the changes to the description of the EDG air start system in the FSAR (and consequently rewords a sentence in the FSAR Safety Evaluation) to remove a sentence stating that there are active redundant components. The change resolves a potential non-conformance between the FSAR description and the way the system is tested. Since plant safety analyses only require redundancy at the level of the emergency AC power sources (the on-site source is supplied by pairs of EDGs), redundant air start systems for each EDG are not required to support the safety design basis of the system. There is no physical change to the plant associated with this activity. This activity is being performed solely to help clarify the basis for the current method of testing the EDGs.

JAF-SE-01-013, REV. 0: **Suppression Pool Monitoring System Digital Upgrade Evaluation**

Activity Type: **N/A**

Suppression Pool Water Temperature instrumentation was replaced and upgraded by modification F1-82-021 in order to meet new requirements set forth in USNRC Regulatory Guide 1.97. Regulatory Guide 1.97 established requirements for design and implementation for post-accident monitoring instrumentation and associated components. Part of this modification was the installation of microprocessor-based components that support monitoring functions for this system. Since these digital components were installed prior to issuance of USNRC Generic Letter 95-02 and NSAC-125, Deviation Event Report (DER 98-01697) was initiated to note that an evaluation of the digital upgrade aspects of this modification should be performed.

The suppression pool water temperature monitoring system evaluated here provides indication and recording of bulk water temperature for post-accident monitoring. This redundant system does not perform any automatic functions, does not interface with any components which can cause an accident initiation or equipment malfunction, and does not perform any automatic functions which could affect any normal effluent release termination functions. Postulated channel failures due to the use of new digital components within these channels will not result in any different types of accidents or malfunctions than those already postulated. Upon a failure of the system, the Operator will remain able to select and view individual temperature devices. This change does not result in an increase in the consequences of an accident or malfunction as evaluated in the FSAR. This change does not result in a design basis limit for a fission product barrier being exceeded or altered. The change does not result in a departure from a method of evaluation described in the FSAR.

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JAF-SE-01-014, REV. 0: **Structural Acceptance Criteria Code Reconciliation (AISC vs ASME) For Torus Attached Piping & SRV Piping Supports**

ACTIVITY TYPE: **Design Change F1-82-020**

The Corrective Action Program identified that the Mark I Plant Unique Analysis Report, TR-5321-2, Rev. 1 and Nuclear Safety Evaluation JAF-SE-83-003, including Nuclear Safety Evaluation JAF-SE-83-003, Addendum 2 and Nuclear Safety Evaluation JAF-SE-83-003, Addendum 3, Rev.2, associated with the Torus Attached Piping (TAP) and Main Steam Relief Lines (SRVs) indicates that pipe supports for modification F1-82-020 were designed (analyzed) per ASME III, Subsection NF, 1977 Edition through Summer 1978 Addenda. A review of pipe support design calculations revealed that they were actually performed to code AISC Manual of Steel Construction, the JAF plant original design code for pipe supports. The use of an alternate evaluation method (i.e., AISC vs. ASME Section III) was evaluated and determined to be acceptable for the Mark I Containment Program (NEDO-24583-1 and NUREG-0661). However, prior NRC approval was not obtained during the implementation of the modification.

This evaluation documents the acceptability of the proposed resolution as supported by report JAF-RPT-MISC-03013, Design Code Discrepancy for Mark I Program Involving Pipe Supports Affected by Modification F1-82-020.

This activity does not degrade below the current design basis the performance of a safety system assumed to function in the accident analysis and does not decrease the reliability of safety systems assumed to function in the accident analysis. This change does not adversely affect the containment, the torus attached piping or the SRV piping performance or reliability in a manner that could lead to an accident or malfunction occurring. This change does not cause the SSCs to be operated outside of their design basis limits. The use of the alternate evaluation method cannot affect any system interface in a way that could lead to an accident and will not result in degradation of safety systems since the affected components have been shown to meet code requirements. Because the impacted equipment and structures have been analyzed using a method comparable to ASME Section III, the response of the equipment and structures will continue to function as designed. Therefore, the activity does not increase the frequency of occurrence or consequences of an accident previously evaluated, does not increase the likelihood or consequences of a malfunction of equipment important to safety previously evaluated, and does not create a different type of accident or a malfunction with a different result than previously evaluated. Additionally, the design basis limit for a fission product barrier as described in the FSAR has not been exceeded or altered. Even though this activity results in the use of a different method of evaluation described in the FSAR (as updated), Section 12.5.1.3, used in establishing the design bases or in the safety analyses and is subject to NRC approval as required by NEDO-24583-1, the activity has been evaluated in the context of 10 CFR 50. 59 and has determined that the methodology used was essentially the same and is not considered a departure from a method of evaluation described in the UFSAR. Therefore, prior NRC approval for this activity is not required.

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JAF-SE-01-015, REV. 1: **GE REM* Light Dryer Wet Transfer System**

Activity Type: **Design Change JD-01-123**

The purpose of this evaluation is to determine if prior NRC approval is required to implement Design Change JD-01-123. This activity will allow use of newly purchased refueling equipment consisting of General Electric REM Light Dryer Wet Transfer System which is comprised of several independent systems that include Main Steam Line Plugs, a Hi-Torque Service Pole system, Guide Rod Extensions, Reactor Flange Protector and Kevlar Separator Lifting Slings. The intent of JD-01-123 is to enhance the defense in depth strategy during a refuel outage by incorporating equipment that decreases the potential for reactor cavity drain down, decrease the reliance on makeup systems, and reduce radiological exposure by allowing early flood up and reducing outage duration.

It is the conclusion of this evaluation that the activities described in JD-01-123 do not require prior NRC approval.

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JAF-SE-02-001, REV. 0: **Evaluate As Installed Sensitivity of Leak Detection Systems For FSAR Update**

Activity Type: **FSAR Change**

This evaluation determines the acceptability of revising the FSAR description of the Leak Detection System (LDS) sensitivity. Licensee Event Report LER-99-013, Rev. 2 documented that the LDS, as installed, could not detect a steam leak of 7 gpm as described in the FSAR as updated. This conclusion was based on preliminary evaluation of the results of engineering calculations JAF-CALC-PC-03300, Revision 1A, JAF-CALC-PC-04051, Revision 0A, JAF-CALC-PC-04074, Revision 0, JAF-CALC-PC-04205, Revision 0, and JAF-CALC-PC-04228, Revision 0. This evaluation reviewed the results of those calculations against the system safety objective and system safety design bases as documented in UFSAR Section 4.10 and determined that while the system as installed did not have the sensitivity described in the UFSAR, the system did meet the safety objective and safety design bases described in the UFSAR. The evaluation evaluated the proposed Page 12 of 13 FSAR changes against the criteria of 10 CFR 50.59 and determined that the changes could be made under this regulation without prior NRC review.

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JAF-SE-02-002, REV. 0:	**Clarification Of Control Room Emergency Ventilation Design And Licensing Basis**
Activity Type:	**Design Change, No. M1-94-195**

This evaluation is being prepared to correct the licensing basis errors and omissions found in Nuclear Safety Evaluation JAF-SE-94-130, Revision 1 supporting design change M1-94-195.

Modification M1-94-195 removed the ability to recirculate air through the Control Room Emergency Ventilation Air Supply System (CREVASS) following an accident by permanently capping the inlet to 70MOD-114. This meant that the post-accident bottled up source term in the Control Room could no longer be radiologically filtered and dilution was needed to reduce the source term concentration. This physical change was potentially adverse since the Control Room operators would most likely receive a higher post-accident dose. The radiological calculations all conservatively assume that no recirculated air passes through the Control Room emergency train and that only outside air passes through the Control Room emergency train. The UFSAR supported the radiological calculations.

Nuclear Safety Evaluation JAF-SE-94-130, Revision 1, in support of M1-94-195, incorrectly stated that "Disabling the modulating capabilities of 70MOD-113 & 114, and capping the inlet to 70MOD-114, will have no effect on the charcoal effectiveness relative to humidity control". This was inaccurate because under conditions when outside humidity is relatively high, the recirculated air formerly lowered the relative humidity of the air entering the CREVASS.

This evaluation concluded that the impact of the proposed activity does not adversely impact licensing or design basis assumption, and prior NRC approval is not required..

JAF-SE-02-003, REV. 1: **Service, Instrument, and Breathing Air Compressor Replacement**

Activity Type: **Design Change M1-99-014**

The purpose of revision 0 of this 10 CFR 50.59 evaluation is to determine if prior NRC approval is required to implement Design Change Package M1-99-014. This activity will replace the existing oil free reciprocating air compressors with oil free rotary screw air compressors.

Revision 1 of this evaluation adds the alternative of using an external monitoring personal computer (PC) in the Relay Room instead of the connection to the EPIC computer. This alternative is required because in the present configuration the EPIC computer does not "Read Holding Registers" or "Preset Single Register" commands as needed by the Intellisys Remoter Interface (IRI) devices.

A thorough review of the FSAR was performed to review the impact of the proposed change. The results of this review determined that the improved reliability of the rotary screw design with improved digital control scheme ensures that there is no more than a minimal increase in the frequency of an accident or likelihood of malfunction of SSCs important to safety. For consequences, the change has no impact on the fuel cladding or related design limits, the reactor vessel or the primary containment including the Main Steam Isolation Valves (MSIVs). As such, the air compressor replacement has no impact on any primary radiological barriers. Additionally, the air systems are not credited with accident mitigation.

With respect to man-machine interface, the FSAR simply indicates that there is Operator control from the Control Room and that the compressors load/unload and auto-start the standby compressor automatically. The start/stop functions will be controlled from the Control Room switches and the compressors will automatically load/unload and start (for standby compressors) based on system pressure. As such, the proposed activity cannot create any accident initiators unique to the new design. No different types of accidents will be created due to human performance issues unique to the new design, since the human-machine interface used for Operator response is similar. The method of analysis used in evaluating the acceptability of the air compressor replacement is based on standard engineering practice and plant specific codes, standards and guidelines, as applicable.

Based on the above, the replacement of the JAF Service, Instrument and Breathing Air Compressors do not require prior NRC review.

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JAF-SE-02-004, REV. 0:	**Technical Requirements Manual, Section 1.1, "Definitions", Discussion of Change L.2**
Activity Type:	**N/A**

In the Safety Evaluation for Improved Technical Specifications (ITS) at FitzPatrick, the NRC approved relocation of some requirements from Technical Specifications to the Technical Requirements Manual (TRM). These relocated requirements are considered a part of the UFSAR and are, therefore, subject to 10 CFR 50.59. Some changes are being made to the relocated requirements. Most changes are administrative in nature. Additionally, some technical changes are being made to the TRM to eliminate unnecessary shutdown requirements, to make the TRM consistent with the ITS, and to provide more accurate or appropriate direction.

This evaluation applies to a technical change, consistent with ITS, being made to the definition CHANNEL FUNCTIONAL TEST for those surveillance requirements relocated to the TRM. This evaluation has determined that the proposed less restrictive changes do not require prior NRC approval.

JAF-SE-02-005, REV. 0: **Technical Requirements Manual, Section 3.0, "TRS/TRO Applicability," Discussions of Changes L.1, L.2, and L.3**

Activity Type: **N/A**

In the Safety Evaluation for Improved Technical Specifications (ITS) at FitzPatrick, the NRC approved relocation of some requirements from Technical Specifications to the Technical Requirements Manual (TRM). These relocated requirements are considered a part of the UFSAR and are, therefore, subject to 10 CFR 50.59. Some changes are being made to the relocated requirements. Most changes are administrative in nature. Additionally, some technical changes are being made to the TRM to eliminate unnecessary shutdown requirements, to make the TRM consistent with the ITS, and to provide more accurate or appropriate direction.

This evaluation applies to three technical changes being made to the 3.0, "Technical Requirements for Operation (TRO) Applicability and Technical Requirements Manual Surveillance (TRS) Applicability" section. This section defines the rules of usage for the TRM Specifications. The three changes eliminate an unnecessary shutdown requirement and make the other rules of usage consistent with those in the ITS. This evaluation has determined that the proposed less restrictive changes do not requires prior NRC approval.

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JAF-SE-02-006, REV. 0:	**Technical Requirements Manual, Section 3.3.B, "Control Rod Block Instrumentation", Discussions of Change L.1 and L.2**
Activity Type:	**N/A**

In the Safety Evaluation for Improved Technical Specifications (ITS) at FitzPatrick, the NRC approved relocation of some requirements from Technical Specifications to the Technical Requirements Manual (TRM). These relocated requirements are considered a part of the UFSAR and are, therefore, subject to 10 CFR 50.59. Some changes are being made to the relocated requirements. Most changes are administrative in nature. Additionally, some technical changes are being made to the TRM to eliminate unnecessary shutdown requirements, to make the TRM consistent with the ITS, and to provide more accurate or appropriate direction.

This evaluation applies to two less restrictive technical changes, consistent with the ITS, being made to the Rod Block specifications relocated to the TRM. This evaluation has determined that the proposed less restrictive changes do not requires prior NRC approval.

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JAF-SE-02-007, REV. 1:	**Technical Requirements Manual 3.3.C, "Post Accident Monitoring Instrumentation", Discussions Of Change L.2, L.3, L.4, L.5, L.6, and L.7**
Activity Type:	**N/A**

In the Safety Evaluation for Improved Technical Specifications (ITS) at FitzPatrick, the NRC approved relocation of some requirements from Technical Specifications to the Technical Requirements Manual (TRM). These relocated requirements are considered a part of the UFSAR and are, therefore, subject to 10 CFR 50.59. Some changes are being made to the relocated requirements. Most changes are administrative in nature. Additionally, some technical changes are being made to the TRM to eliminate unnecessary shutdown requirements, to make the TRM consistent with the ITS, and to provide more accurate or appropriate direction.

This evaluation applies to less restrictive changes being made to the Post Accident Monitoring Instrumentation relocated to the TRM. Revision 0 of this evaluation characterized the changes discussed in this evaluation as administrative in nature. Revision 1 eliminates that characterization and evaluates the changes on the basis of technical merit. This evaluation has determined that the proposed less restrictive changes do not requires prior NRC approval.

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JAF-SE-02-008, REV. 0: **Technical Requirements Manual, Section 3.3.N, "Automatic Depressurization System (ADS) Inhibit", Discussion Of Change L.1**

Activity Type: **N/A**

In the Safety Evaluation for Improved Technical Specifications (ITS) at FitzPatrick, the NRC approved relocation of some requirements from Technical Specifications to the Technical Requirements Manual (TRM). These relocated requirements are considered a part of the UFSAR and are, therefore, subject to 10 CFR 50.59. Some changes are being made to the relocated requirements. Most changes are administrative in nature. Additionally, some technical changes are being made to the TRM to eliminate unnecessary shutdown requirements, to make the TRM consistent with the ITS, and to provide more accurate or appropriate direction.

This evaluation applies to a change to the existing requirement for the testing of the inhibit function of ADS. This evaluation has determined that the proposed change does not require prior NRC approval.

JAF-SE-02-009, Rev. 1: **Technical Requirements Manual 3.4.A, "Structural Integrity", Discussion Of Change L.1**

Activity Type: **N/A**

In the Safety Evaluation for Improved Technical Specifications (ITS) at FitzPatrick, the NRC approved relocation of some requirements from Technical Specifications to the Technical Requirements Manual (TRM). These relocated requirements are considered a part of the UFSAR and are, therefore, subject to 10 CFR 50.59. Some changes are being made to the relocated requirements. Most changes are administrative in nature. Additionally, some technical changes are being made to the TRM to eliminate unnecessary shutdown requirements, to make the TRM consistent with the ITS, and to provide more accurate or appropriate direction.

This evaluation applies to a technical change being made to the Structural Integrity specification that eliminates a mandatory shutdown requirement and replaces it with actions consistent with Generic Letter 91-18, "Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions". Revision 0 of this evaluation characterized the changes discussed in this evaluation as administrative in nature. Revision 1 eliminates that characterization and evaluates the changes on the basis of technical merit. This evaluation has determined that the proposed less restrictive change does not requires prior NRC approval.

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JAF-SE-02-010, REV. 0: **Final Feedwater Temperature Reduction**

Activity Type: **Design Change JD-02-122**

A mode of operation called Final Feedwater Temperature Reduction (FFTR) will be used at FitzPatrick to extend operation beyond normal end of cycle (EOC) operation. After the normal EOC is reached, the reactor will continue to produce energy but at a lower power. This mode of operation is called coastdown. The purpose of this proposed activity is to maintain a higher power level than would normally be achieved during coastdown conditions. Higher thermal power is achieved by the positive reactivity addition of colder feedwater to the reactor. While the overall thermal efficiency of the plant is reduced, there is a net benefit of additional electric generation. This Final Feedwater Temperature Reduction change will allow more electric generation when coastdowns are necessary.

This evaluation has determined that the proposed activity does not require NRC approval prior to implementation.

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JAF-SE-02-011, REV. 0: **Cycle 16 Core Reload**

Activity Type: **Design Change JD-01-102**

This evaluation supports the design change package for the Cycle 16 core reload and documents safety analyses performed on the Cycle 16 core and the thermal limits established in the Core Operating Limits Report (COLR), Revision 14. The core will be loaded with 196 fresh GE14 fuel bundles. This is a new fuel type for the JAF core. JD-01-102 contains a description of the fuel and core for Cycle 16. Cycle 16 specific analyses were performed according to the methodologies detailed in the General Electric Standard Application for Reload Fuel. The COLR was revised to include limits for GE14 fuel, the specific reload fuel bundles, and for the core as a whole. Ten control rod blades (CRB) will be replaced by new MARATHON CRBs, and other CRBs will be shuffled to limit their exposure. FSAR changes are required to support the new fuel introduction, changes to the Power-Flow map (Exclusion Region), and cycle-specific transient calculations.

The analyses demonstrate that the core as proposed can be operated safely within the limits specified in the COLR. The frequency and consequences of accidents, transients and malfunctions are unchanged by the Cycle 16 core reload. No new accidents or malfunctions of a different type were identified. No design basis limits were altered or exceeded. There is no departure from methods of evaluation described in the FSAR.

Attachment II

Summary of Plant Changes, Test, and Experiments for 2001/ 2002 as Required by 10 CFR 72.48

Entergy Nuclear Operations, Inc
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

ATTACHMENT II
Summary of Plant Changes, Test, and Experiments for 2001/2002 as Required by 10 CFR 72.48

Introduction

10 CFR 72.48 (c) (1) states in part:

> A licensee or certificate holder may make changes in the facility or spent fuel storage cast design as described in the final safety analysis report, make changes in the procedures as described in the final safety analysis report, and conduct tests or experiments not described in the final safety analysis report.....

if the change , test or experiment does not meet any of the criteria in paragraph (c) (2) of this section.

10 CFR72.48 (d) (2) states in part:

The licensee and certificate holder shall submit, as specified in § 72.4, a report containing a brief description of any changes, tests, and experiments, including a summary of the evaluation of each.

Unless otherwise noted, each 10 CFR 72.48 evaluation listed concluded that it's subject change, test, or experiment did not:

- Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the final safety analysis report;
- Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the final safety analysis report;
- Result in more than a minimal increase in the consequences of an accident previously evaluated in the final safety analysis report;
- Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the final safety analysis report;
- Create a possibility for an accident of a different type than any previously evaluated in the final safety analysis report;
- Create a possibility of a malfunction of an SSC important to safety with a different result than previously evaluated in the final safety analysis report;
- Result in a design basis limit for a fission product barrier as described in the final safety analysis being exceeded or altered; or
- Result in a departure from a method of evaluation described in the final safety analysis report used in establishing the design bases or in the safety analyses;

and therefore, did not require prior NRC approval prior to implementation.

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JAF-ISFSI-01-001, REV. 0: **HI-STORM JAF Overpack**

Activity Type: **Design Change**

The proposed changes in this evaluation are intended to provide modifications to an as licensed HI-STORM 100 overpack to allow for the overpack to be moved into (empty) and moved out (loaded) of the FitzPatrick reactor building railroad access doors. This will permit the transfer of a loaded multi-purpose canister (MPC) from the HI-TRAC to occur within the secondary containment providing additional levels of environmental protection for the cast loading campaign.

The HI-STORM FitzPatrick overpack meets the design specification criteria, and the proposed changes herein do not decrease any safety margins or safety factors below acceptable limits. Specifically, the shielding, structural, thermal, radiological protection, operations and accidents were examined in detail. Therefore, the HI-STORM FitzPatrick overpack is safe to use as designed.

JAF-ISFSI-01-002, REV. 0: **ECO 1024-30, Revision 2**

Activity Type: **Design Change**

The purpose of this evaluation is to review proposed changes to the HI-STORM FSAR. Specifically, revise Section 1.D.4 to specify fabrication sequencing to allow access to the overpack body and pedestal cavity for placing concrete, clarifying the use of non-shrink grout, delete the redundant description of placing concrete in the HI-STORM lid, and change the concrete compressive strength, maximum aggregate size, and air content. Additionally, revise HI-STORM FSAR Appendices 3.A, 3.D, 3.G, and 3.M to make adjustments to analyses in which the concrete compressive strength was a factor.

This evaluation focuses on the elements of the change that cause analyses contained in the HI-STORM FSAR to be revised. The change has no effect on any criticality, containment, thermal, or shielding analysis contained in the FSAR.

The elements of concern are the changes to analyses caused by the concrete compressive strength being reduced. In each corrected analysis when available strength is compared to the strength required by NUREG or other design standards, the resulting Safety Factor is greater than 1. The changes meet applicable NRC design, material, and construction requirements as determined by correcting the same analyses employed to prove the validity of the original design.

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JAF-ISFSI-01-003, REV. 0: **ECO 1025-2, Revision 1**

Activity Type: **Design Change**

The purpose of this evaluation is to review proposed changes to the HI-TRAC 125 and Transfer Lid design drawings. The proposed changes are being made to improve fabrication activities, improve or enable fit up of mating components, correct drawing errors, and incorporate design enhancements resulting from lessons learned in manufacturing the prototype HI-TRAC 125 and Transfer Lid. Some of the changes have effects on structural analyses. The structural components with design characteristics that are factors in structural analyses are determined by calculation in the revised analyses to have Safety Factors greater than one. The cask design changes affect the HI-TRAC 125 with serial No.002 and the associated Transfer Lid.

The evaluation focuses on the elements of the change that cause structural analyses contained in the CFSAR to be revised. The change has no effect on any critical, containment, thermal, or shielding analysis contained in the CFSAR.

The elements of concern are the elimination of the tongue from the HI-TRAC 125 Bottom Flange and the corresponding groove from both the Pool Lid and the Transfer Lid. To compensate for the resulting loss in the ability to prevent sideways movement of the Pool Lid and Transfer Lid in case of a side drop accident, the number of bolts for mounting the lids on the HI-TRAC Bottom Flange was increased from 18 to 24. The use of additional bolts in lieu of fewer bolts and a mating tongue-and-groove joint to oppose shear forces is evaluated. In each case that available strength is compared to strength required by NUREG or other design standards, the resulting Safety Factoris greater than 1. The change meets applicable NRC, design, material, and construction requirements as determined by revising the same analyses employed to prove the validity of the original design.

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JAF-ISFSI-01-004, REV. 0: **ECO 1025-4, Revision 1**

Activity Type: **Design Change**

The purpose of this evaluation is to review proposed changes to the HI-STORM FSAR and to HI-TRAC 125 and Transfer Lid design drawings. The HI-STORM FSAR is being revised to remove a weld efficiency factor, taken in error from the ASME Code, Subsection NG, which is not required by the controlling Subsection NF. Drawing revisions incorporate changes to correct typographical errors, increase the width of the Transfer Lid top and bottom plates, to change selected full penetration welds to partial penetration welds, and to reduce the size of selected partial penetration welds. The proposed change also corrects analyses affected by the design changes of ECO 1025-4, Revision 1.

This evaluation focuses on the elements of the change that cause structural analyses contained in the HI-STORM FSAR to be revised. The change has no effect on any criticality, containment, thermal, or shielding analysis contained in the HI-STORM FSAR.

The elements of concern are the changes of weld types and sizes that are factors in analyzing the structural adequacy of the transfer cask and associated lids. Evaluating the structural adequacy of the affected welds is addressed in attachment B to ECO 1025-4, Revision 1. In each case that available strength is compared to the strength required by NUREG or other design standards, the resulting Safety Factor is greater than 1. The change meets applicable NRC, design, material, and construction requirements as determined by revising the same analyses employed to prove the validity of the original design.

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JAF-ISFSI-01-005, REV. 0: **ECO 1025-7, Revision 0**

Activity Type: **Design Change**

The purpose of this evaluation is to review proposed changes to the HI-STORM FSAR and to HI-TRAC 125 Transfer Lid design drawings. The HI-STORM FSAR is being revised to correct an analysis affected by the design changes of ECO 1025-7, revision 0, specifically change 1 ½ inch and 1 inch full penetration welds to 0.75 inch and 0.50 inch groove welds, respectively.

This evaluation focuses on the elements of the change that cause the stress analysis of the Transfer Lid housing components contained in the HI-STORM FSAR to be revised. The change has no effect on any criticality, containment, thermal, or shielding analysis contained in the HI-STORM FSAR.

The elements of concern are the changes of weld types and sizes that join components of the Transfer Lid housing which are factors in a stress analysis. In each case that available strength is compared to the strength required by NUREG or other design standards, the resulting Safety Factor is greater than 1. The change meets applicable NRC, design, material, and construction requirements as determined by revising the same analyses employed to prove the validity of the original design.

Attachment III

Summary of Regulatory Commitment Changes for 2001/2002

Entergy Nuclear Operations, Inc
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

ATTACHMENT III
Summary of Regulatory Commitment Changes for 2001/2002

2001 Change No. 1

Commitment:

Contained in NRC Inspection Report 90-004, were results of reviewed problem areas within the Emergency Service Water (ESW) System. Included were a listing of FitzPatrick corrective actions taken to resolve these issues. Included was: "...To ensure adequate cooling to the safety-related loads, NYPA isolated ESW cooling to non-safety loads except for the CRD pumps (7gpm) and the recirculation pumps (275 gpm). Procedures were already in place to isolate ESW to the recirculation pumps if an ESW injection occurred. As a result, all safety-related components could be supplied with adequate ESW flow, provided operators isolated the flow to the recirculation pumps.

Source Document:

NRC Region 1 Inspection Report No. 50-333/90-04, dated 08/06/1990

Revised Commitment:

ESW isolation to non-safety loads no longer required.

Justification For Change:

Engineering Design document JD-99-015 documents the ESW hydraulic analysis which demonstrates adequate cooling capacity for required safe shutdown loads during a postulated Appendix R event.

ATTACHMENT III
Summary of Regulatory Commitment Changes for 2001/2002

2002 Change No. 1.

Commitment:

Perform final LP inspection of CRDRL nozzle blend radius and vessel wall during the refuel outage for Cycle 12;

OR

The next time in-vessel penetrant inspection of the feedwater nozzle/sparger is required by NUREG-0619, Table 2 .

To compensate for the extended period until final LP, the Authority proposes to perform a television visual inspection (less than 1 ml resolution) of the nozzle blend radius and adjacent vessel wall during each refuel outage until final LP inspection is performed

Source Documents:

- Fitzpatrick letter to NRC (JPN-83-64), CRD Return Line Modification (NUREG-0619), dated 07/07/1983.
- NRC letter to FitzPatrick, CRD Return Line Modification, dated 08/25/83.
- FitzPatrick letter to NRC (JPN-85-65), NUREG-0619 BWR Feedwater Nozzle and CRD Return Line Nozzle Cracking, dated 08/20/1985.

Nozzle NDE Actions FitzPatrick has taken Since 1985:

- 1987 Refuel outage – Performed VT-1,
- 1988 PERFORMED UT INSPECTION,
- All refuel outages up to 1998 (RF-13) – performed VT-1
- Refuel outage 14(fall 2000)-Performed Enhanced Visual Examination EVT-1, (1/2 mil resolution).

(NOTE: No indications were noted during these NDE examinations)

Revised Commitment:

(1) Eliminate the requirement to perform a VT-1 visual examination of the CRD nozzle blend radius and adjacent vessel wall area during each refuel outage, and eliminate the final LP inspection in the same area.

AND

(2) The EVT visual examination of the nozzle blend radius and adjacent vessel wall area will be performed every 10 year interval beginning in 2000.

2002 Change No. 1. (Continued)

Justification For Change:

(1) Several non-destructive examination methods have been performed to date, all resulting in the CRDRL nozzle blend radius having no recordable or relevant indications. These examinations were all performed after the CRDRL nozzle was cut and capped (during the 1983 refuel outage), and included the necessary modifications required by the selected option during the 1985 refuel outage. The used methods include: (a) by UT in 1988; (b) by VT-1 since 1987 refuel outage, until 1998 refuel outage (RF13); and (c) by EVT-1 in RF14 (fall 2000). The enhanced visual examination (EVT-1) offers a better resolution to detect surface flaws than the previous used VT-1;

(2) The effectiveness of the EVT-1 exam performed in RF14 showed no relevant indications in the CRDRL nozzle blend radius and adjacent vessel wall area. EVT-1 is the preferred visual examination method by the BWRVIP for flaw detection;

(3) The absence of potential thermal fatigue concern issues since the CRDRL was permanently cut and capped.

(4) Eliminating the need to perform LP examinations of the CRDRL nozzle blend radius and adjacent vessel wall area will avoid unnecessary radiation exposure to plant personnel, and potential damage to in-vessel components, without compromising plant safety;

(5) It is important to note that the previous commitment of LP examination for all four feedwater nozzles has been eliminated by FitzPatrick letter (JPN-94-031) to the NRC, dated 07/19/1994.

(6) The change in frequency and in inspection methodology although different from the previous commitment, continues to provide reasonable assurance of the structural capability of the SSC to perform its intended function.

2002 Change No. 2

Commitment:

In response to NRC Generic Letter 90-03,"Vendor Interface For Safety–Related Components", FitzPatrick stated that ..."The FitzPatrick vendor manual control procedures include a checklist to provide additional assurance that the proper technical information updates are being provided. However, these procedures do not require periodic contact with the vendors of key safety-related equipment.

The Authority will revise and implement changes to the FitzPatrick vendor manual control procedures to require annual contact with vendors of key, safety-related, non-NSSS equipment....".

Source Documents:

- NRC Generic Letter 90-03, "Relaxation of Staff Position in Generic Letter 83-28, Item 2.2 Part 2, Vendor Interface For Safety Related Components."
- FitzPatrick letter to NRC (JPN-90-066), "Response to Generic Letter 90-03, vendor Interface For Safety-Related Components", dated 09/28/1990.

Revised Commitment:

Entergy Nuclear Northeast (ENN) will revise and implement changes to the FitzPatrick's vendor manual control procedures to require contact with vendors of key safety-Related non-NSSS equipment on a recurring three year cycle.

Justification for Change:

NIRMA 90-03 white paper suggests vendors on the recontact list should be contacted on a three year rotating basis. The results of a survey conducted by NIRMA in 2000 state the majority of utilities use a three year recontact period.

2002 Change No. 3

Commitment:

In an updated response to NRC Generic Letter 89-13, concerning test programs to verify the heat transfer capability of safety-related heat exchangers cooled by service water, specifically, Emergency Diesel Generator Heat Exchangers, FitzPatrick stated that ..."in lieu of testing, two of the four EDG heat exchangers are opened each refueling outage for visual and eddy current inspection."

Source Documents:

- NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," dated 07/18/1989.
- FitzPatrick's Updated Response (JPN-93-015) to NRC Generic Letter 89-13, dated 03/16/1993.

Revised Commitment:

Emergency Diesel Generator heat exchanger inspections are being performed with the same periodicity but are being done during major EDG LCOs rather than being restricted to performance during refueling outages.

Justification For Change:

The intent of the original commitment, to perform EDG heat exchanger inspections on regular intervals, is being maintained. The revised schedule for the inspections allows for greater flexibility and use of plant resources.

2002 Change No. 4

Commitment:

In an updated response letter to NRC Bulletin 96-13, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors", FitzPatrick committed to the following: The torus (suppression pool) will be desludged every other refueling outage commencing with refueling outage #15 (Fall 2002), based upon engineering calculations and observed sludge generation rate..

Source Documents:

- FitzPatrick letter (JAFP-00-0288) to the NRC, "Completion of Actions for NRC Bulletin 96-03, Potential Plugging of ECCS Suction Strainers, dated 12/11/2000.
- NRC Letter to FitzPatrick, "James A. FitzPatrick Nuclear Power Plant – Completion of Licensing Action for Bulletin 96-03, Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors", dated 02/08/2001.

Revised Commitment:

Torus will be evaluated for desludge activities prior to each refueling outage commencing with R16 (fall 2004).

Justification For Change:

An Engineering "Strainer Performance Analysis" was completed to evaluate, in part, the torus sludge accumulation rate based on the most recent torus cleanup results. The evaluation results supported conclusions that torus desludge activities were not required during the fall 2002 refueling outage.

2002 Change No. 5

Commitment:

In correspondence with the NRC to update the Commission on the status of the plant's Pipe Support Inspection Program, in response to NRC Bulletin 79-02, Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts", NRC Bulletin 79-07, "Seismic Stress Analysis of Safety-Related Piping", and NRC Bulletin 79-14, "Seismic Analysis For As-Built Safety-Related Piping Systems", FitzPatrick stated that "...The Authority will coordinate the pipe support rework task with the FitzPatrick Inservice Inspection (ISI) Program. The pipe support rework task will include engineering evaluations, revisions to design drawings or the repair ...of deficiencies. Drafting and engineering personnel will...complete the rework. This combined effort will occur over the next six refueling outages."

Source Document:

- FitzPatrick Letter (JPN-91-020) to the NRC, "Long Term Pipe Support Inspection and Evaluation Program", dated 05/03/1991.

Revised commitment:

Completion of the Pipe Support Inspection Program activities will occur by December 31, 2003.

Justification For Change:

All outage related Pipe Support Inspection Program work was completed during the plant's fall 2002 refueling outage (RF15). The remaining non-outage Pipe Support Inspection Program work will be completed by the end of 2003. This extension has no adverse affect on nuclear safety.

2002 Change No. 6

Commitment:

In letter JPN-85-87, FitzPatrick responded to NRC safety Evaluation Report that addressed both the Inservice Test Program (IST) for Pump and Valves for the first ten year interval. Included in this response letter were expanded Program testing requirements for High Pressure Coolant Injection (HPCI) System components. Specifically, the HPCI "...turbine exhaust line vacuum breakers will be exercised with each IST test of the HPCI turbine. Closure will be verified by the LLRT performed during each refueling outage."

Source Document:

- FitzPatrick letter (JPN-85-87) to the NRC, "Response to Safety Evaluation Report for Inservice Testing (IST) Program for the 1st Ten Year Interval", dated 09/27/1985.

Revised Commitment:

Turbine exhaust line vacuum breaker testing will be performed in accordance with Fitzpatrick's IST Program, revision 6, 3rd Interval Test Plan as approved by the NRC.

Justification For Change:

This commitment was superceded. HPCI vacuum breaker testing is being performed in accordance with the IST Program Plan as accepted by the NRC per JAF Technical Specifications T.S.) Amendment No. 234 and Improved Technical Specifications (ITS) Safety Evaluation Report.

2002 Change No. 7

Commitment:

NUREG-0619 outlined the NRC staff's proposed resolution of Generic Technical Activity A-10, feedwater nozzle cracking due to bypass flow. FitzPatrick, in response, adopted numerous recommendations of NUREG-0619. Subsequent to these implemented recommendations, FitzPatrick requested, and was granted a change to a NUREG-0619 commitment that eliminated the need to perform a liquid penetrant test (PT) examination of the feedwater nozzle. As part of the justification, FitzPatrick committed to utilize the Leakage Monitoring System (LMS) to detect bypass flow.

Source Documents:

- FitzPatrick letter (JPN-94-031) to the NRC, "NUREG-0619 Inspection Program for Feedwater Nozzles", dated 07/19/1994.
- Fitzpatrick letter (JPN-99-003) to the NRC, "Feedwater Nozzle Leakage Monitoring System", dated 02/18/1999.
- NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking", dated 11/1980.

Revised Commitment:

FitzPatrick will discontinue utilizing of LMS to detect feedwater bypass flow.

Justification For Change:

The bases for the acceptability of the commitment change includes the effectiveness of the hardware changes implemented in response to NUREG-0619 and improvements in UT examination techniques.

2002 Change No. 8

Commitment:

Included in an October 29, 2001 10 CFR 50.72 event notification (EN#38554) to the NRC addressing an "Invalid Primary and Secondary Containment Isolation Due to Loss of Power to Reactor Protection System Power Distribution Bus B", FitzPatrick reported that the event cause was due to an invalid trip signal caused by an Electrical Protective Assembly (EPA) logic card failure. Also included in the notification were actions taken and/or actions to be taken to prevent recurrence. Specifically, ...New capacitors will be installed on the logic cards, or refurbished cards will be installed in the subject EPAs during the next available maintenance window.

Source Document:

- FitzPatrick's 10 CFR 50.72 Event Notification (Work Sheet No. 38554) to NRC, dated 10/29/2001, "Invalid Primary and Secondary Containment Isolation Due to Loss of Power to Reactor Protection System Power Distribution Bus B".

Revised Commitment:

Three EPA's require capacitor/logic board replacement:
Replace EPA logic cards, or refurbish with new capacitors, 71EPA-RPS1A2G, 71EPA-RPS1B1T and 71EPA-RPS1B2T. The remaining 5 EPAs are in acceptable condition in terms of capacitor life.

Justification For Change:

The original service information letter (General Electric SIL 496) advised licensees that the capacitor be replaced in 29 years if used at less than 35° C (95° F) ambient, placing the scheduled replacements by 2008. A review of the eight EPA's in terms of capacitor replacement determined 5 EPAs to be acceptable in terms of capacitor life. Of the remaining 3 EPA logic cards, one card refurbishment was completed in March 2003, the second and third cards are scheduled for refurbishing in July and October 2003 respectively.

2002 Change No. 9

Commitment:

A corrective action from FitzPatrick LER-00-009 stated that " ...The EQ equipment qualification Document Reports (QDRs) are upgraded in accordance with an EQ Improvement Plan which predated this event. During the QDR upgrade process, the corrective actions from the previous review of Generic Letters, Information Notices, Bulletins and Circulars are reviewed for accuracy. If a corrective action is determined to be inadequate, the operating experience document is reopened for review."

"The QDR up grade process will therefore evaluate the response to NRC generated operating experience documents relevant to environmental qualification.
(QDR upgrade process is scheduled to be completed July 1, 2002)"

Source Document:

- Licensee Event Report LER-00-009 (Letter No. JAFP-00-0216), "HPCI and A&B Core Spray Systems Inoperable Due to Lack of Proper Environmental Qualification on Minimum Flow Valve Control Circuits", dated 09/22/2000.

Revised Commitment:

The QDR upgrade process was scheduled to be completed by July 1, 2002. Based on reviews completed to date, no additional EQ issues have been identified. The QDR upgrade process will continue to completion in accordance with the EQ improvement Plan schedule. (Complete)

Justification For Change:

Scheduled Engineering resources will continue to complete the reviews in accordance with the EQ Improvement Plan. The EQ Group, with assistance of EPRI, NUS and NUGEQ, has prepared report JAF-RPT-MISC-04000, "Environmental Qualification (EQ) Evaluation of Applicable NRC Information Notices, Bulletins, Generic Letters, and Circulars." This report identifies the NRC correspondence that potentially affect the EQ Program at FitzPatrick and further identifies the NRC correspondence that affect each individual EQ Qualification File.

2002 Change No. 10

Commitment:

In NRC Notice of Violation 95-03-01, FitzPatrick was cited for failure to maintain a controlled inspection control program for portable radiation monitors. In response to this Notice of Violation, Corrective Actions To Be Taken, included "...Radiation Protection instrument procedure RP-INST-103, "Issue of Radiological Equipment", is being updated to include the instrument inventory check on a weekly basis."

Source Documents:

- NRC Inspection Report No. 50-333/95-03 and Notice Of Violation, dated 02/03/1995.
- FitzPatrick Letter (JAFP-95-0122) to the NRC, "Response to Notice of Violation NRC Inspection Report 50-333/95-03", dated 03/06/1995.

Revised Commitment:

Radiological equipment, including inservice, issued and available for issue will be verified on a monthly basis for proper location and appropriate response checks in accordance with radiation Protection procedure RP-OPS-02.06, "Control and Issue of Radiological Equipment".

Justification For Change:

Weekly inventory checks performed during the past six years have not shown a repeated similar problem as identified in NRC Notice of Violation 95-03-01. Subsequent NRC and INPO inspections also have not identified similar problems, and computer software upgrades (equipment tracking programs) provide RP technicians daily listing and due dates of instruments requiring response checks.

ATTACHMENT III
Summary of Regulatory Commitment Changes for 2001/2002

2002 Change No. 11

Commitment:

A corrective action from FitzPatrick Licensee Event Report (LER) 86-001 stated that "...Prior to or during plant startups two independent audits will be performed to ensure all Operations Department surveillance tests are properly scheduled."

Source Document:

- Licensee Event Report LER-86-001 (Letter No. JAFP-86-0256), "Failure To Perform Surveillance Test At Required Frequency", dated 03/24/1986.

Revised Commitment:

Commitment has been deleted. The Surveillance Test Program is controlled within an Administrative Procedure whose purpose includes surveillance procedures that have an event based test interval.

Justification For Change:

Plant Administrative Procedure AP-19.01, "Surveillance Testing Program" has successfully provided the necessary controls to assure surveillance tests, which are reactor mode dependent, are properly scheduled and performed at their require frequencies/intervals.