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JAFP-05-0069

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

SUBJECT: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Summary of Plant Changes, Tests, and Experiments for 2003 and 2004 as Required by 10 CFR 50.59 AND 10 CFR 72.48, and Summary of Commitment Changes for 2003 and 2004

Dear Sir:

This letter transmits the summary of changes, tests and experiments implemented at the James A. FitzPatrick Nuclear Power Plant (JAF) for the years 2003 and 2004 as required by 10 CFR 50.59(d)(2) and 10 CFR 72.48(d)(2). Also included is the summary of revised regulatory commitments as required 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, "Managing Regulatory Commitments Made by Power Reactor Licensees to the NRC Staff."

Attachment 1 provides the summary of each 10 CFR 50.59 report including a brief description of the change, test, and experiment, and a summary of the evaluation of each. Also included is the JAF assigned 10 CFR 50.59 evaluation number (e.g., JAF-SE-03-001), report revision number (if applicable), title, activity type, and engineering change number (if applicable).

Attachment 2 provides the summary of each regulatory commitment change requiring NRC notification, and a brief statement of the basis for the change. Also included is the JAF tracking number of the change, and the basis document from which the commitment was made.

There were no required 10 CFR 72.48 reports generated during 2003 and 2004.

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

Very truly yours,

Kevin Mulligan
Site Vice President (Acting)

T.A. Sullivan
Site Vice President

TAS:GB

Attachments: 1. Summary of 10 CFR 50.59 Reports for 2003 and 2004
2. Summary of Regulatory Commitment Changes for 2003 and 2004

cc: Regional Administrator
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ATTACHMENT 1
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Summary of 10 CFR 50.59 Reports for 2003 and 2004

James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Entergy Nuclear Operations, Inc

Summary of Changes, Tests, and Experiments for 2003 and 2004

Introduction to the 2003 and 2004 10 CFR 50.59 Report

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:**
> **a change to the technical specifications...is not required,**
> **the change, test, or experiment does not meet any of the criteria in paragraph (c)(2) ...**

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

> **The licensee shall maintain records of changes ... made pursuant to paragraph (c) of this section. These records must include a written evaluation which provides the bases for the determination that the change, test, or experiment does not require a license amendment ...**

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. A report must be submitted at intervals not to exceed 24 months.**

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

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

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JAF-SE-95-018, REV. 1: **Drywell Entries During Startup And Shutdown**

ACTIVITY TYPE: **Procedure Change**

This 50.59 evaluation determined the acceptability of allowing Drywell entries during power operation, with reactor conditions of (1) less than or equal to 10 percent rated thermal power, (2) reactor mode switch not in Run, and (3) a controlled reactor power level with no planned evolutions which could result in plant power changes. This evaluation is required because FSAR Section 5.2.3 currently restricts Drywell entry during power operations. This evaluation provides the basis to change the FSAR to allow Drywell entry provided the above restrictions are followed. Controls have been established through procedure implementation to ensure the health and safety of workers entering the Drywell environment.

Revision 1 of the 50.59 evaluation is being performed to support revising statements in the FSAR regarding drywell entry limitations. This change will allow Drywell access for power levels at 15% or less of rated thermal power. This evaluation will provide the controls required to be in place to assure reactor power remains ≤ 15% power during Drywell entry. This evaluation is required as a result of proposed changes to plant procedures.

The presence of personnel in the Drywell does not alter the design or operation of plant systems. Primary Containment Integrity will be maintained at all times while personnel are accessing the Drywell. Therefore, entry of personnel into the Drywell during the plant conditions specified above is acceptable.

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JAF-SE-01-015, REV. 2: **GE REM Light Dryer Wet Transfer System**

ACTIVITY TYPE: **Engineering Change (No. JD-01-123)**

Revision 2 addressed the concerns of Condition Report CR-JAF-2002-05095 which stated that 10CFR50.59 questions A3 through A7 with respect to the Hi-Torque Service Pole System, Guide Rod Extensions, Reactor Flange Protector and Kevlar slings were not adequately answered. The above named equipment are associated with refueling activities and are part of the GE REM Light Dryer Wet Transfer System.

Revision 2 enhanced the responses to questions A3 through A7 with respect to the Hi-Torque Service Pole System, Guide Rod Extensions, Reactor Flange Protector and Kevlar slings by addressing additional potential failure mechanisms and the corresponding consequences.

JAF-SE-03-001, REV. 0: **UFSAR Update To Incorporate Revised MSLBA & CRDA Radiological Accident Consequences**

ACTIVITY TYPE: **UFSAR Update**

The proposed activity is to revise the UFSAR to incorporate recent updates to the JAF Design Basis Accident (DBA) dose consequence analyses, specifically the Main Steam Line Break Accident (MSLBA) and Control Rod Drop Accident (CRDA). These DBAs were re-analyzed to incorporate changes to some of the input parameters and assumptions used in the radiological analyses. This information supports a revision to the UFSAR and resolves outstanding open items associated with the UFSAR. The appropriate calculations were revised as supporting documentation.

JAF-SE-03-002, REV. 0:	**Updated Reactor Pressure Vessel Fatigue Analysis**
ACTIVITY TYPE:	**Engineering Analysis and UFSAR Update**

The original Reactor Pressure Vessel (RPV) fatigue analysis will be updated based on plant operating data. The number of cyclic events will be adjusted to reflect actual plant data. Based on this evaluation, UFSAR Table 4.2-3 will be revised to reflect the allowable number of cycles for each cyclic event.

The RPV fatigue evaluation is based on the same ASME code requirements as the original fatigue evaluation. The maximum projected 60-year cumulative usage factor is less than 1.0, which is the ASME code limit. Fatigue critical RPV components were determined and appropriate fatigue evaluations were performed. These cyclic events will be monitored by JAF procedure.

JAF-SE-03-003, REV. 0: **Elimination Of SRV Accumulator Check Valve Leakage Testing Per Surveillance Test (ST)-39M**

ACTIVITY TYPE: **Procedure Change**

Leakage testing of ADS SRV Accumulator Check Valves, performed via a pressure drop test of each accumulator and associated piping, was instituted in the 1980s to demonstrate the capability of the accumulators to support required short-term ADS actuation requirements. Subsequently, upgrades to the Drywell pneumatic supply subsystem provided a reliable, safety-related supply of nitrogen to the SRV actuators under all postulated conditions. This upgraded supply, in conjunction with the pneumatic capacity of the accumulators, is able to satisfy the short-term and long-term ADS pneumatic requirements, regardless of check valve leakage. The upgraded configuration is such that no single active failure or accident consequence will depressurize the pneumatic supply or will prevent supplying the required pneumatic pressure to safety-related pneumatic components. Therefore, check valve leakage testing is no longer required, and will be eliminated. External leakage of the check valves or leakage elsewhere in the pneumatic system is detectable by other means. Therefore, elimination of check valve leakage testing has no impact on the transient initiation potential or the transient and accident mitigation functions of the SRVs.

JAF-SE-03-004, REV. 0:	**Reduction Of The Sample Number Of Welds Requiring Inspection Under The Main Steam & Feedwater Augmented Inspection Program**
ACTIVITY TYPE:	**3rd Ten-Year Inservice Inspection Program and Technical Requirements Manual (TRM) Change**

This 50.59 evaluation supports a revision to the 3rd Ten-Year Inservice Inspection (ISI) Program and the Technical Requirements Manual (TRM) Surveillance Requirement (TRS) 3.4.A.2, involving the inspection of high stressed circumferential piping joints in the main steam and feedwater lines. The TRM requires a 100% volumetric inspection of the welds per inspection interval. The evaluation determined the acceptability of reducing the inspection requirement of TRS 3.4.A.2 from a 100% volumetric inspection to a reduced percentage, and used a selection process including criteria such as risk significance categories of high and medium, and those welds with the highest stress values and multiple degradation mechanisms.

The main steam sample inspection population will be reduced from 12 welds to 2 welds in the 3rd Ten-Year ISI Program and subsequent ISI intervals. The feedwater sample inspection population will be reduced from 22 welds to 4 welds in the 3rd Ten-Year ISI Program and subsequent ISI intervals.

JAF-SE-03-005, REV. 0: **EHC Scram Frequency Reduction**

ACTIVITY TYPE: **Engineering Change (No. JD-03-019)**

This engineering design change was processed to provide turbine circuit enhancements to reduce the susceptibility to and the frequency of scrams at the James A. Fitzpatrick Nuclear Power Plant. These changes were based on General Electric (GE) Technical Information Letter (TIL) No. 1212-2.

This 50.59 evaluation specifically addresses and reviews the elimination of the main turbine high Exhaust Hood Temperature (EHT) trip aspect of this design change. The design change reduces the likelihood of a turbine trip due to the failure of a single instrument. The results of this review determined that the removal of the EHT trip was acceptable based on redundant turbine trips and procedural controls. The removal of this trip is enveloped by the current FSAR analysis that addresses the potential failure of the turbine generator and the impact on the control room and other vital safety features of the plant.

ATTACHMENT 2
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Summary of Regulatory Commitment Changes for 2003 and 2004

James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Entergy Nuclear Operations, Inc

2003 Change No. 1

Commitment Source Document:

FitzPatrick response letter (No. JAFP-80-675, dated 08/27/1980) to NRC Bulletin No. 80-17, Supplement 3, Failure of Control Rods to Insert During a Scram at a BWR.

Commitment:

The above reference contained the following statement: The FitzPatrick plant has implemented administrative controls which require an immediate manual scram in the event of multiple control rod drift alarms or the presence of a marked change in the number of control rods which exhibit high temperature alarm conditions.

Revised Commitment:

Remove the administrative requirement to manually scram the reactor when 5 or more control rod drives have high temperature alarms.

Justification For Change:

The plant commitment, as referenced above, was performed as an interim measure to prevent control rod failure (to fully insert during a scram) due to hydraulic locking. As long term corrective measures to this issue, the NRC issued Generic SER, BWR Scram Discharge System, dated 12/01/1980, which outlined long-term design and performance criteria that BWRs had to meet to address the Scram Discharge Volume (SDV) header issues. FitzPatrick has since completed design changes (modifications) to the SDV header which resolved the long-term issues. Therefore, the interim measure is no longer applicable.

2003 Change No. 2

Commitment Source Document:

FitzPatrick letter to the NRC (No. JPN-93-015, "Updated Response to Generic Letter 89-13, Service Water System Problems Affecting Safety Related Equipment," dated 03/16/1993).

Commitment:

Included in the above reference, Section 4.2, concerning the heat transfer capability of safety-related heat exchangers, was a statement that in lieu of performance testing of the Emergency Diesel Generator (EDG) heat exchangers, "two of the four EDG heat exchangers are opened every refueling outage for visual and eddy current inspections."

Revised Commitment:

Two of the four EDG heat exchangers are opened every refueling cycle for visual and eddy current inspections.

Justification For Change:

This commitment was made in a time frame when scheduled EDG maintenance was performed during refueling outages (24 month operating cycle). FitzPatrick has since transitioned to on-line EDG maintenance. The scheduled activities remain on the same committed 24 month frequencies.

2003 Change No. 3

Commitment Source Document:

NRC Event Notification Report No. 38554, dated 12/10/2001, Invalid Primary and Secondary Containment Isolation Due to Loss of Power to Reactor Protection System Power Distribution Bus B.

Commitment Change Request No. 2002-008, dated 10/2002.

Commitment:

Following the above event, troubleshooting activities identified a failed capacitor located within the failed circuitry. Corrective actions listed on the Event Notification Worksheet stated that new capacitors (or refurbished logic cards) will be installed.

JAF Commitment Change No. 2002-008 (10/2002) revised the corrective action to state that boards will be replaced as refurbished replacement parts become available.

Revised Commitment:

An Engineering review determined that vendor refurbishment activities consisted of replacing capacitors on the circuitry logic cards. Availability of materials and expertise on site supported that JAF would conduct the logic card refurbishment activities in lieu of the vendor.

Justification For Change:

This change remains consistent with the intent of the original corrective actions submitted to the NRC in Event Notification Report No. 38554, dated 12/10/2001.

2003 Change No. 4

Commitment Source Document:

FitzPatrick letter to the NRC (No. JPN-84-58, "Qualification of ADS Accumulators, NUREG-0737 Item II.K.3.28," dated 09/04/1984).

Commitment:

In response to an NRC letter requesting additional information regarding the qualification of the Automatic Depressurization System (ADS) accumulators at JAF, the following commitment was made: A test of ADS accumulator check valves will be performed before the first startup after each refueling outage to assure that leakage from these valves are within acceptable limits.

Revised Commitment:

The commitment has been revised to eliminate check valve testing as originally described based on 10 CFR 50.59 Evaluation No. JAF-SE-03-003, rev. 0, "Elimination of SRV Accumulator Check Valve Leakage Testing per ST-39M."

Justification For Change:

Leakage testing of ADS SRV Accumulator Check Valves was instituted in the 1980s to demonstrate the capability of the accumulators to support required short-term ADS actuation requirements. Upgrades to the Drywell pneumatic supply subsystem provided a reliable, safety-related supply of nitrogen to the SRV actuators under all postulated conditions. Therefore, check valve leakage testing is no longer required.

2003 Change No. 5

Commitment Source Document:

FitzPatrick letter to the NRC (No. JPN-91-020, "Long-Term Pipe Support Inspection and Evaluation Program," dated 05/03/1991) concerning NRC Bulletins 79-02, 79-07, and 79-14.

Commitment Change Request No. 2002-005, dated 09/2002.

Commitment:

The above reference reported that JAF's Pipe Support Inspection Program was completed. Additionally, JAF reported that, in response to the results of the completed inspections, pipe support rework task, involving: (1) engineering evaluations; (2) revisions to design drawings; or (3) repairs of less significant deficiencies that did not impact operability, would be coordinated with the Inservice Inspection Program. The effort to complete the rework task would occur over the next six refuel outages (up to and including refuel outage #15, Fall 2002).

Commitment Change Request No. 2002-005 (dated 09/2002) was generated which extended the completion date of the rework task to December 2003.

Revised Commitment:

During refuel outage #15, all pipe support field work requiring a plant outage was completed. All remaining non-outage related work associated with the pipe support rework task effort will be completed by December 2005.

Justification For Change:

The extension of the completion date of the remaining engineering work associated with the pipe support evaluations and design drawing revisions will not impact operability of the supports nor impact the operability of the respective piping systems. The above tasks are of low safety significance as they do not impact the ability of the supports to perform their safety function.

Similarly, it was determined that the scheduled extension dates for repair/restoration of the remaining supports will not impact operability of the supports or impact the operability of the respective piping systems. The nonconforming conditions are considered minor.

2004 Change No. 1

Commitment Source Document:

As part of JAF's Technical Specifications (TS) Amendment No. 253, which approved the extension of the Allowed Outage Time (AOT) for the Emergency Diesel Generators (EDGs) from 7 days to 14 days, the NRC relied on several commitments discussed in the NRC Safety Evaluation Report (SER).

Commitment:

One of these commitments involved making the second EDG of the inoperable EDG subsystem "available" for manual operation during days 8 through 14 of the 14-day AOT.

Revised Commitment:

Both EDGs of an EDG subsystem may be made inoperable and unavailable for manual operation for all or any portion of the 14 days allowed by the TS AOT, provided the risk is assessed and managed in accordance with plant procedures that implement 10 CFR 50.65(a)(4).

Justification For Change:

As discussed in the SER for TS Amendment No. 253, the NRC relied on other factors in addition to the above commitment. These included the plant having controls in place to reduce the likelihood of risk-significant plant configurations during the AOT, and the plant having implemented a risk-informed Configuration Risk Management Program to assess the risk associated with the removal of equipment from service during the AOT.

This change has an insignificant impact on the overall level of safety of the plant and on the safety functions of systems, structures, and components.

2004 Change No. 2

Commitment Source Document:

A 1991 NRC Diagnostic Evaluation Team Inspection of JAF's Fire Protection Program identified a weakness concerning the assigning of only one individual to walk-down the plant part time for transient combustibles.

Commitment:

JAF responded that there would be documented independent weekly supervisory /management tours of the plant whose purpose would include the monitoring of transient combustibles.

Revised Commitment:

The commitment has been withdrawn.

Justification For Change:

Currently, Fire Protection inspections of the plant are performed under Fire Protection procedures. These inspections are performed primarily by the Fire Protection and Safety Department staff and are periodically supported by the Fire Protection and Safety Coordinator. All safety related areas are inspected on a weekly basis, with non-safety related areas inspected monthly. Discrepant conditions are entered into the Corrective Action program for identification and resolution.

Based on the administrative controls currently in place at JAF for control of combustible materials, control of hot work and ignition sources, and impairment tracking of out-of-service Fire Protection systems, the weekly supervisory/management inspections are no longer required.

2004 Change No. 3

Commitment Source Document:

Following the completion of a Diagnostic Evaluation Team (DET) inspection in 1991, the NRC staff requested additional information regarding resolution of the DET identified issues. Specifically, design deficiencies were noted involving the Emergency Diesel Generator (EDG) air start system.

Commitment:

Develop a procedure to open air receiver bank isolation valves when one EDG air compressor is out of service to allow the in-service compressor to charge both banks of the EDG air receivers.

Revised Commitment:

The commitment to align both banks of the EDG air receivers to the remaining (in-service) EDG air compressor when taking one EDG air compressor out of service was withdrawn.

Justification For Change:

The EDG Safety Design Bases located in the FSAR states that the EDG must be capable of automatic start at any time. The EDG air system supports this Safety Design Bases. The commitment change does not prevent the EDG from automatically starting since one starting air bank is kept in service. Each EDG air start system is capable of supplying sufficient air for 10 starts. The system consists of 2 banks of receivers, each capable of supplying air for 5 starts. The banks can be lined up independently or cross connected. The Technical Specifications LCO for Diesel Fuel Oil, Lube Oil, and Starting Air, requires that the applicable subsystems be within the required limits. The Technical Specifications Bases states that the starting air system is required to have a minimum capacity for five successive EDG starts without recharging or realigning the air start receivers. This change maintains one bank of receivers capable of meeting the requirement for 5 successive EDG starts.

2004 Change No. 4

Commitment Source Document:

A 1982 NRC Emergency Preparedness Appraisal (Inspection) identified an NRC concern (question #14) which stated, "Provide assurance that the necessary respiratory protection is available for personnel in the EOF and that self contained breathing apparatus (SCBAs) will be available for each emergency team." FitzPatrick responded by letter to the NRC (No. JPN-82-67, dated 8/23/82) and made the commitment described below.

Commitment:

"... SCBAs have been budgeted for 1983. They will be purchased and made available for all the EOF emergency teams by 7/30/83."

Revised Commitment:

The commitment has been withdrawn.

Justification For Change:

The removal of the commitment does not impact or lesson the ability of the emergency team members to perform emergency response functions. The emphasis or philosophy in 1982 was to "prevent internal exposure at uptake at all costs." However, with the 1992 revision to 10 CFR 20, the emphasis or philosophy was changed with the industry now aligned with a new focus on overall dose reduction. Additionally, at the time the commitment was made, the EOF was located at an on-site location. The new EOF location is now outside the 10-mile Emergency Planning Zone. Appropriate cartridge respirators for removal of particulates and iodines are available for EOF field teams should the situation warrant.

2004 Change No. 5

Commitment Source Document:

In response to the NRC's request for additional information regarding the adequacy and availability of design bases information, FitzPatrick committed in a letter to the NRC (No. JPN-97-010, dated 3/10/97) to implement an initiative to identify and correct Final Safety Analysis Report (FSAR) discrepancies. Included was the following:

Commitment:

Implement a program of periodic vertical-slice assessments at both Indian Point and FitzPatrick. A minimum of one assessment per operating cycle will be conducted at each plant.

Revised Commitment:

The commitment has been withdrawn.

Justification For Change:

The purpose of these assessments was to confirm the design bases contained in the FSAR were properly maintained following completion of the initiatives to review, identify and correct FSAR deficiencies. These assessments are no longer required to assure FSAR adequacy since processes initiated after the commitment was made have been successful in controlling the FSAR. Included in these processes are administrative procedures dealing with FSAR update preparation and control and current licensing basis deviations. The update process is consistent with the guidance provided in NEI 98-03, "Guideline for Updating Final Safety Analysis Reports." The success and rigor of the revised update process make these assessments unnecessary.

2004 Change No. 6

Commitment Source Document:

A routine NRC inspection in 1990 (NRC Inspection Report No. 50-333/90-08, dated 2/13/91) identified a weakness regarding JAF not entering LCOs for primary containment isolation valves (PCIVs) rendered inoperable during surveillance testing. The existing policy was to not enter the applicable LCOs for systems or components made inoperable during surveillance testing. This position was reconsidered and JAF committed to enter the LCO and comply with the requirements whenever a surveillance test intentionally rendered a PCIV inoperable.

Commitment:

JAF committed to enter the LCO and comply with the requirements whenever a surveillance test intentionally rendered a PCIV inoperable. Interim guidance was distributed pending implementation of the required procedure changes.

Revised Commitment:

This commitment is no longer required to ensure operation in accordance with the plant's Technical Specifications (TS) and is withdrawn.

Justification For Change:

In August 2002, JAF converted from its current TS to the Improved Standard TS (ITS). The ITS clearly require the above action when a PCIV is rendered inoperable. In addition, the general operating philosophy has changed since the early 1990's. Site administrative procedures dealing with maintenance activities during LCOs, LCO tracking, and the conduct of operations all provide examples of this operating philosophy change. These procedures give clear guidance in terms of entering, tracking and complying with TS Actions, regardless of whether there is a commitment associated with the activity. Therefore, this commitment is outdated and is no longer necessary.

2004 Change No. 7

Commitment Source Document:

Corrective Action No. 1, contained in Licensee Event Report (LER) 90-011, rev. 0, "Shutdown Cooling Isolation – Deficient Procedure Allowed Reverse Flow of Normal Starting Pressure Transient to Trip Isolation Pressure Switches."

Commitment:

Corrective Action No. 1 states in part "Temporary changes were made to Operating Procedure OP-13, "Residual Heat Removal System", under the section for "Shutdown Cooling Configurations" to require stopping of the reactor recirculation pumps and closing of the pump discharge valves prior to starting the RHR pumps for shutdown cooling. This change is in accordance with recommendations of NSSS supplier and effectively isolates the pressure sensor from the initial pressure surge. In addition, a caution was added to note that spurious high suction pressure isolations may occur when starting the B side RHR system."

Revised Commitment:

The commitment is revised to allow a single Residual Heat Removal (RHR) pump in one piping loop to be run simultaneously with both Reactor Water Recirculation (RWR) pumps running in slow speed.

Justification For Change:

The revised operating conditions for the RWR System and RHR System (shutdown cooling mode) were evaluated and approved by the NSSS supplier under General Electric Report No. GE-NE-0000-0018-1418-04-R0, dated April 2004 (Operability Report on Reactor Pressure Vessel Temperature Control During NobleChem Re-Application Following Reactor Shutdown at James A. FitzPatrick Nuclear Power Plant). The combined RHR and RWR flow per loop is limited to 24,000 gpm. The reason for the original corrective action not to run RHR and RWR pumps simultaneously was the concern for water hammer sensed at high reactor pressure instrumentation causing RHR shutdown cooling isolation. Subsequent plant experience and evaluation determined that venting of the pressure instrument sensing lines was inadequate which lead to collapsed voids and the resultant RHR shutdown cooling isolations. Specific venting guidance has been proceduralized and the pressure transients from an RHR pump start should not result in an RHR shutdown cooling isolation. The revised operating allowance will permit proper deposition of noble metals and greater flexibility in operating the plant during shutdown and heat-up evolutions.

2004 Change No. 8

Commitment Source Document:

Corrective Action No. 8, contained in Licensee Event Report (LER) 91-006, rev. 1, "Manual Reactor Shutdown Due to Inoperability of Both Low Pressure Coolant Injection Subsystems Due to Mechanical Failure of One Valve in Each of the Two Systems."

Commitment:

Valves 10MOV-27A/B (Residual Heat Removal (RHR) System, Low Pressure Coolant Injection (LPCI) Loop A & B, Outboard Injection Throttle and Primary Containment Isolation Valves) are being placed in the full open position to reduce vibration during use of the Shutdown Cooling (SDC) Mode.

(NOTE: This corrective action was taken following discovery that 10MOV-27B sustained stem damage due to vibration induced cyclic fatigue caused by excessive throttling during the SDC mode of RHR operation.)

Revised Commitment:

10MOV-27A/B may be throttled, provided fluid velocity through the valves is not allowed to exceed 2400 feet per minute at any time.

Justification For Change:

During injection of noble metals, the RHR System is placed in the SDC mode of operation. In order to achieve optimum deposition of noble metals, RHR flow rate must be maintained at approximately 7000 gpm, which requires system throttling. The original commitment to maintain 10MOV-27A/B fully open in the SDC mode was placed in effect in response to a component failure as documented in the above LER. The failure was attributed to vibration induced cyclic fatigue. An engineering evaluation was performed and established that, based on valve manufacturer input, limited throttling with valve fluid velocity $\leq$ 2400 feet per minute will not result in a similar failure. Since the revised commitment places limitations on the use of 10MOV-27A/B for throttling, the original cause of the failure continues to be addressed.