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May 15, 2007
JAFP-07-0065

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

SUBJECT: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
License No. DPR-59
**Summary of Plant Changes, Tests, and Experiments for 2005 and 2006 as Required by 10 CFR 50.59 and 10 CFR 72.48, and Summary of Commitment Changes for 2005 and 2006**

Dear Sir or Madam:

This letter transmits the summary of changes, tests and experiments implemented at the James A. FitzPatrick Nuclear Power Plant (JAF) for the years 2005 and 2006 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, a summary of each evaluation, the JAF assigned 10 CFR 50.59 evaluation number (e.g., JAF-SE-05-001), report revision number (if applicable), title, activity type, and engineering change number (if applicable).

Attachment 2 provides the summary of each 10 CFR 72.48 report including a brief description of the change, test, and experiment, a summary of each evaluation, the JAF assigned 10 CFR 72.48 evaluation number (e.g., JAF-ISFSI-05-001), report revision number (if applicable), title, and activity type.

Attachment 3 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 (e.g., CCR-05-001), 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. Jim Costedio, Regulatory Compliance Manager, at (315) 349-6358.

Very truly yours,

Pete Dietrich
Site Vice President

PD:JC:ed

Attachments: 1. Summary of 10 CFR 50.59 Reports for 2005 and 2006
2. Summary of 10 CFR 72.48 Reports for 2005 and 2006
3. Summary of Regulatory Commitment Changes for 2005 and 2006

cc: Mr. Samuel Collins
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**ATTACHMENT 1**
**JAFP-07-0065**

# Summary of 10 CFR 50.59 Reports for 2005 and 2006

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

# ATTACHMENT 1
# JAFP-07-0065
## 2005 Summary

### JAF-SE-05-001, Rev. 0: ASME Code Repair of Containment (Torus) JAF-05-24673

### ACTIVITY TYPE: Design Change

Nuclear Change ER JAF-05-24673 implemented a repair to restore the integrity of the Torus in accordance with original design requirements; therefore, there were no adverse effects on nuclear safety. Furthermore, enhanced consideration of fatigue loading from the HPCI exhaust not considered in the original design or Mark I Program improvements were included in the nuclear change.

### JAF-SE-05-002, Rev. 0: Evaluation of the Capability of Cooling Water Intake Bar Heaters

### ACTIVITY TYPE: Evaluation

The activity expanded on UFSAR Section 12.3.7 by clarifying the performance capability of the non-safety related QA Category II/III bar heaters. The intake heaters are not important to safety since minimum required flows to safety-related pumps can be maintained whether or not the intake bar heaters are preventing frazil ice accretion (build-up) on the intake bars. The intake heaters are needed for normal full power operation and not for the safe shutdown of the plant. Engineering Report 3971 indicates that the safety-related flow of 11,700 gpm (assuming post-LOCA conditions with one ESW pump and two RHRSW pumps) does not cause plant icing problems due to increased "residence" time of the water in the intake tunnel, the very small magnitude flow field and the size of the intake "entrainment" zone is too small to draw ice particles down to the intake structure, and ice that is accreted on the intake structure bars will be eroded at high water velocities which will allow some flow to pass through the hardened ice. Engineering Report 3971 shows that if all the intake bar heaters fail, sufficient intake flow area will be maintained to provide cooling flow for the safety pumps; therefore, there is no change in consequences associated with describing actual intake bar heater capability. UFSAR Section 12.3.7 page 12.3-6 describes the capability of the intake bar (deicing) heaters. UFSAR section 12.3.7 page 12.3-7 states that deicing heaters are not required for the safe shutdown of the plant. These two sections together indicate that the intake bar heaters are not required. ESW and RHRSW flows will continue to be capable of providing design flows under accident conditions. Frazil ice accretion on the intake bars is not a malfunction with a different result, since the result is bounded by that previously evaluated in UFSAR Section 12.3.7 page 12.3-7. The fission product design basis limits as described in the UFSAR are not affected by the proposed activity because adequate cooling flow will be available when a frazil ice condition exists to maintain the UFSAR described fission product barrier design basis limits. The UFSAR does not describe the method of evaluation used to establish the design basis of the intake bar heaters.

**JAF-SE-05-003, Rev. 0: Shutdown Cooling Using Main Steam Line Drains and the Main Condenser**

**ACTIVITY TYPE: Procedure Change**

The proposed activity involved use of Temporary Operating Procedure (TOP) 354 to provide shutdown cooling using the main steam line drains and the main condenser. This was necessitated by finding a through wall crack on the common Residual Heat Removal (RHR) shutdown cooling suction line. The evaluation considered the impact of heat removal on the structures, systems, and components (SSCs) involved, not the total heat removal capacity of the proposed operating mode. Heat removal capacity was required to be evaluated for the existing conditions per Technical Specification (TS) 3.4.8 and its bases prior to depending on this mode alone for alternate decay heat removal.

Accidents/transients normally associated with condenser issues are associated with the steam path from the reactor and are precluded by the plant shutdown conditions. Potential for condenser shell failure due to water loading was minimal because the shell had been demonstrated to be capable of carrying the water loads required. Potential for gross failure was low due to the low energy of the water systems involved. Consequences of gross tube failure were limited by sampling of the hotwell and Circulating Water System (CWS); hotwell activity was maintained below ODCM effluent limits, and any detectable CWS activity was required to be evaluated and corrected. Based on these factors TOP-354 was approved.

## 2006 Summary

In 2006 no 10 CFR 50.59 Evaluations were approved. JAF-SE-03-003 was withdrawn per memorandum JENG-06-0009. The withdrawal was due to an inadequacy in the evaluation identified during the 2005 10 CFR 50.59 Inspection and documented as a Non-cited Violation in NRC Inspection Report 05000333/2005006.

ATTACHMENT 2
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# Summary of 10 CFR 72.48 Reports for 2005 and 2006

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Docket No. 50-333
Entergy Nuclear Operations, Inc

# ATTACHMENT 2
# JAFP-07-0065
## 2005 Summary

**JAF-ISFSI-05-001, Rev. 0: JAF-RPT-SFS-04329 Revision 2, JAF ISFSI 10 CFR 72.212 Evaluation Report**

**ACTIVITY TYPE: Evaluation**

The as-load condition of MPC Serial Number 014 and 015 is an analyzed condition and the CoC LCO 3.1.1 Required Action B.2 has been met. The design function of the MPC helium backfill is to ensure adequate heat transfer during storage and provides an inert atmosphere for long-term fuel integrity. The reduced helium does not adversely affect the structural stress and the confinement boundary leak rates as internal pressures are reduced. The thermal analysis results computed temperatures of the as-loaded condition and the HI-STORM 100 System FSAR are essentially the same, and are within the margin of accuracy of the method of analysis. The lower helium density condition does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the CFSAR/SFSAR. The lower helium density condition does not 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 CFSAR/SFSAR. The potential for an increase in consequences is not impacted by the results of the thermal analysis for the credible scenarios evaluated in the CFSAR. As such, the consequences of these events are bounded by the original CFSAR analyses. Since the lower helium density in the MPC does not adversely affect the structural stress nor the confinement boundary leak rates, and the component temperatures remain essentially the same, the change does not result in more than a minimal increase in the consequences of a malfunction of an SSC previously evaluated in the CFSAR/SFSAR. The change does not introduce any new materials or conditions that would create a new accident scenario that has not been previously analyzed. Therefore, the lower helium density does not create the possibility of an accident of a different type than previously evaluated in the CFSAR/SFSAR. All existing off-normal events and accidents potentially affected by thermal performance have been previously analyzed and bound the change in this evaluation. Therefore, this change does not create the possibility for a malfunction of an SSC important to safety with a different result than previously evaluated in the CFSAR/SFSAR. The lower helium density maintains an inert environment for long term integrity of the stored fuel and there are no adverse effects on the thermal stress levels, pressure stress levels, and confinement leakage for the affected components. Therefore, the lower helium density in the MPC's does not result in a design basis limit for a fission product barrier as described in the CFSAR/SFSAR from being exceeded or altered. The analysis by Holtec to evaluate the effects of the lower helium density was the same as the original analysis methodologies and does not result in a departure from a method of evaluation described in the CFSAR/SFSAR used in establishing the design basis or in the safety analysis.

## 2006 Summary

In 2006 no 10 CFR 72.48 Evaluations were performed.

**ATTACHMENT 3**
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# Summary of Regulatory Commitment Changes for 2005 and 2006

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

# 2005 Summary

## CCR-05-001

***Commitment Source Document(s):***
"Response to Deficiency B", Letter No. JAFP-78-0347, dated 7/11/1978; JAF Response to NRC Violation (Report No. 88-01), Letter No. JAFP-88-0384, dated 4/25/1988; JAF Letter to NRC, "Commitment Change Regarding Surveillance On Locked Valves, Letter No. JAFP-98-0319, dated 10/1/1998

***Commitment:***
Perform locked valve surveillance semiannually and prior to startup from a refueling outage.

***Revised Commitment:***
The commitment is withdrawn. This withdrawal does not apply to the requirement to perform the locked valve surveillance required by the Fire Protection Program.

***Justification For Change:***

Surveillance history has shown that the present administrative controls (i.e., improved equipment status control and SOMS tagging program) have proven effective in maintaining equipment in the desired configuration. Therefore, the ability of an SSC to perform its intended safety function is not affected by the change.

## CCR-05-002

***Commitment Source Document:***
Letter JPN-91-020

***Commitment:***
Complete Engineering, drafting, and repair of pipe supports over the next six refueling outages beginning in 1991 (i.e., up to and including RO-15).

***Revised Commitment:***
All remaining non-outage work associated with the pipe support rework task will be completed by December 2006.

***Justification For Change:***
The remaining pipe supports scheduled for repair / restoration will not impact the operability of the supports or impact the operability of the respective piping systems.

# 2006 Summary

## CCR-06-001

***Commitment Source Document:***
Administrative Procedure AP-05.02, Control of Temporary Modifications, NRC Unresolved Item (URI) 91-01-04

***Commitment:***
Revise AP-05.02 to require treating changes to fire doors and ventilation systems as potential temporary modifications.

***Revised Commitment:***
This commitment was withdrawn for the JAF commitment tracking system.

***Justification For Change:***
The procedure revision was not a commitment as defined by the commitment management process. This item was tracked in the JAF procedure as a commitment, however, review of NRC Inspection Report 92-27 which closed URI 91-01-04 showed that the URI was closed based on improved performance not the procedure enhancement. The control of temporary modifications will continue to be controlled via procedure (EN-DC-136) but this item does not require tracking as a commitment.

## CCR-06-002

***Commitment Source Document:***
Letter JPN-89-061 attachment I paragraph X.c.7

***Commitment:***
Instrument Air Testing – A test program for header isolation valves will be implemented for the service air and breathing air to close at a set pressure on loss of air.

***Revised Commitment:***
This commitment is withdrawn.

***Justification For Change:***
JPN-89-061 provided response to Generic Letter 88-14. The intent of Generic Letter 88-14 was "...verification by test that air operated safety related components will perform as expected in accordance with all design basis events...". As document in memorandum JREG-06-008 the service and breathing air header isolation valves do not have and do not support any safety functions associated with the compressed air system. Therefore the commitment for surveillance testing per Generic Letter 88-14 is not required.

# ATTACHMENT 3
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## CCR-06-003

***Commitment Source Document:***
Letter JAFP-80-0672/LER 80-065, Letter JAFP-97-0322/ LER 97-008

***Commitment:***
Check Standby Gas Treatment System drain line quarterly for evidence of excessive silt and corrosion build-up.

***Revised Commitment:***
Check Standby Gas Treatment System drain line semi-annually for evidence of excessive silt and corrosion build-up. If after 2 years of semi-annual performances of the drain line check the results have been satisfactory then the frequency should be extended to annual.

***Justification For Change:***
Engineering Request JAF-05-22563 evaluated the results of quarterly testing since 1997 and determined that extending the testing interval to semi-annual was warranted. The evaluation also established acceptance criteria that if met over 4 consecutive performances of the semi-annual testing would provide basis for extending the frequency further.

## CCR-06-004

***Commitment Source Document:***
Letter JAFP-97-0322/ LER 97-008

***Commitment:***
Develop a periodic Preventive Maintenance task to periodically flush the Standby Gas Treatment System drain line.

***Revised Commitment:***
This commitment is withdrawn.

***Justification For Change:***
Flushing the Standby Gas Treatment System drain line is only required if excessive build-up is identified in semi-annual inspection of the drain line. Therefore, a separate commitment and task to perform periodic flushing is not required.

## CCR-06-005

***Commitment Source Document:***
NUREG 0737, Information Notice 84-51

***Commitment:***
A licensed operator will perform a second verification of the return to normal of safety related systems for OPs, STs, protective tags, and jumper installation/removal. ST qualified I&C personnel will perform second verifications on I&C STs.

***Revised Commitment:***
A program is in place to ensure properly trained and knowledgeable personnel perform verification of the return to normal of safety related systems.

***Justification For Change:***
Due to the increase in training, qualification, accreditation, and industry standards over the past 27 years, a licensed operator is no longer the only individual capable of performing an independent verification and understanding the adverse implication of a mispositioned component or loss of status control.

## CCR-06-006

***Commitment Source Document:***
Letter JPN-91-020

***Commitment:***
Complete Engineering, drafting, and repair of pipe supports by December 2006.

***Revised Commitment:***
All remaining non-outage work associated with the pipe support rework task will be completed by December 2007.

***Justification For Change:***
The remaining pipe supports scheduled for repair / restoration will not impact the operability of the supports or impact the operability of the respective piping systems.