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October 28, 1996
JAFP-96-0429

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

Subject: **James A. FitzPatrick Nuclear Power Plant**
Docket No. 50-333
Supplementary Information on the Plant Shutdown Experienced by the James A. FitzPatrick Nuclear Power Plant on September 16, 1996

References:

1. JAF LER-96-010, Dated; October 16, 1996; Plant Shutdown Due to Human Error Inadvertently Connecting Two Terminals While Calibrating Protective Relay
2. JEP-96-089; Review of Potential Off Hours Emergency Plan Implementation for the September 16, 1996 Notice of Unusual Event (NUE)
3. JDED-APL-96-017; Action Plan for Design Review of Plant Event (Rupture Disk) of September 16, 1996
4. James A. FitzPatrick Technical Specifications
5. Failure Evaluation of "D" RHR (10640) 4KV Circuit Breaker Contact Failure (Transmitted via JMD-96-425) [Failure date: October 16, 1996]
6. JAF LER-96-002, Dated; April 12, 1996; Potential Common Mode Failure of Circuit Breakers in Both Safety Divisions Due to Design or Installation Error
7. Technical Specification Amendment 207; Elimination of The Main Steam Line Isolation Valve Closure and Reactor Scram Function of The Main Steam Line Radiation Monitor
8. JAF-RAS-94-004; Evaluation of Potential Deviation Associated With Technical Specification Amendment 207
9. PSA Applications Guide; EPRI TR-105396, August 1995

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Dear Sir:

On September 16, 1996 the James A. FitzPatrick (JAF) Nuclear Power Plant experienced a plant trip initiated by human error. The details of this event were reported in accordance with 10CFR50.73(a)(2)(iv) in Licensee Event Report (LER) 96-010 (reference 1). The plant response to this event, as well as the causes and corrective actions associated with this event, have been the topic of a number of discussions between the plant staff and the NRC staff.

The Region 1 NRC staff requested that additional information be provided, in the form of docketed correspondence, to address NRC questions asked during these conversations. These specific NRC questions are addressed below:

1. **Q:** Could the [JAF] Plant Staff have effectively responded to this event if the event had occurred on the "off-hours" when supplementary staff was not immediately available?

   A: Yes; The JAF Plant Staff conducted a review of essential activities required to be conducted in approximately the first two hours of the event. The activities considered in this review are outlined below:

   - Execute operating procedures (including emergency and abnormal operating procedures)

   - Execute the security plan

   - Execute the emergency plan

   The review considered the staff available to conduct these activities for the time period between the event initiation and the arrival of support personnel called out via the emergency plan. The review made use of the simulator to model operator actions to review task sequencing and durations for approximately the first hour and twenty minutes of the event. This review is documented in internal correspondence (reference 2) and considered operator actions as well as support function tasks. The review determined that:

   a. In the time period prior to activation of the emergency facilities, some tasks may have been delayed and the prioritization of tasks for plant operators, security and radiological protection personnel would have been different, however, the requirements of the activities identified above could have been satisfied.

   b. Actions that may have been delayed would not have resulted in additional problems and would not have led to an escalation in Emergency Plan classification.

   c. JAF commitments regarding plant staff activation and response could have been met for this event if it had occurred "off-hours".

2. Q: Does actuation of the LP Turbine and RFPT turbine rupture disks represent a [JAF] design inadequacy?

A: **Under Evaluation**; The transient response of plant systems is undergoing detailed analysis to verify design adequacy. A preliminary evaluation will be completed by December 1, 1996 (JAF commitment No. JAFP-96-0411-03 from LER-96-010, reference 1). The technical approach to be employed for this evaluation is documented in internal correspondence (reference 3) which identifies specific actions required to support this analysis.

This evaluation is identifying all relevant plant design criteria. The plant design is being evaluated relative to these criteria to determine adequacy. The evaluation is also analyzing the plant response to abnormal operational transients and design basis accidents (relevant to the September 16, 1996 plant trip) to determine design adequacy. This analysis is evaluating the ability to satisfy the applicable safety design bases and demonstrate that postulated events do not result in unacceptable consequences for abnormal operational transients and design basis accidents as defined in the JAF Updated Final Safety Analysis Report (UFSAR).

The analysis above has identified a potential deviation from the JAF Updated Final Safety Analysis Report (UFSAR) associated with Technical Specification Amendment 207 (reference 7). A Reasonable Assurance of Safety (RAS) (reference 8) was prepared to evaluate the acceptability of continued plant operation until the plant shutdown scheduled for October 26, 1996 for refuel outage 12 (RFO 12). The RAS concluded that there was a reasonable assurance that continued plant operation, until RFO 12, does not constitute a threat to the health and safety of the public. The RAS was approved by the Plant Operating Review Committee on October 23, 1996.

The JAF staff has informed the resident inspectors and will continue to inform the resident inspectors on the status of this analysis.

3. Q: What were the cause and corrective action taken for the failure of the "D" RHR pump to start from the control room switch when attempting to put this pump in service for torus cooling?

A: **Cause:** This failure was discussed in JAF-LER-96-010 (reference 1) which stated that an equipment failure evaluation (EFE) (reference 5) was performed on the failure of the "D" RHR pump circuit breaker to close. The EFE concluded that switch contacts within the circuit breaker closure mechanism intermittently failed to make contact at a consistent contact point due to some looseness in one of the switch's stationary contacts. This caused intermittent failure of the close coil to energize when the switch exhibited sufficiently high contact resistance. This switch was used in an application where the switch was breaking approximately six amperes of inductive current. It is postulated that the failure experienced on September 16, 1996 is due to a combination of the switch application and internal contact geometry.

Corrective Action: A different failure cause for these switches was experienced previously at JAF as reported in JAF LER-96-002, Dated; April 12, 1996 (reference 6). Engineering evaluation performed as a consequence of the event described in LER-96-002 has determined that these contacts can be electrically bypassed (jumpered out). Based on this analysis, a modification was prepared to electrically bypass these contacts. This modification has now been performed on all safety related 4KV circuit breakers which require automatic or manual closure to perform their intended accident mitigation function. There are no other switches of this design used on 4KV circuit breakers which require automatic or manual closure to perform their intended accident mitigation function.

Because of previous experience with failures of these 4KV circuit breakers, operating experience documentation relevant to failure of these switches was re-reviewed to determine applicability. For the documents reviewed, there were no industry events that were attributed to a loose terminal on this type of switch. Information Notice (IN) 95-02 identified manufacturing flaws in these switches. The batch date code and manufacturing facility identifier for the failed switch is not one of those referenced in IN 95-02 and the failed switch did not exhibit any of the manufacturing flaws discussed in IN 95-02.

LER-96-002 described a failure cause for these switches, where the switch contact resistance was a function of switch cycles. The failure cause evaluated in the reference 5 EFE (event described in LER 96-010 Dated; October 16, 1996) described the switches ability to achieve and maintain contact-to-contact continuity to be a function of contact point geometry as determined by internal switch tolerances.

LER-96-002 stated that the observed defect resulted in a "substantial safety hazard" as defined in 10CFR21 and therefore, the NRC Emergency Operations Center was informed of this determination on March 21, 1996. As corrective action for LER-96-002, The contact blocks in safety related circuit breakers with a history of more than 1,500 close cycles were replaced prior to April 12, 1996.

The switch failures experienced on September 16, 1996 were due to the same failure mode (loss of contact-to-contact continuity) as the failures reported in LER-96-002 but had a different cause (contact point geometry vs. switch cycles). The failures identified in LER-96-002 were characterized as potential common mode failures (potentially effecting both safety system trains), the failure evaluated in the reference 5 EFE has not been determined to be a potential common mode failure because it has not been determined if the internal switch tolerances observed are specific to this failure or are the result of a (switch) design or manufacturing condition.

This switch has been returned to the switch manufacturer for further analysis. This analysis will determine if the switch failure was due to a design or manufacturing condition and is due to be complete by November 14, 1996. This failure evaluation will be used to determine if the failure observed on September 16, 1996 constitutes a "substantial safety hazard" as defined in 10CFR21. This determination will be made by November 21, 1996.

Commitment: The switch evaluated in the reference 5 EFE has been returned to the switch manufacturer for further analysis. This analysis will determine if the switch failure was due to a design or manufacturing condition and will be complete by November 14, 1996. This information will be used to determine if the condition could constitute a "substantial safety hazard" as defined in 10CFR21. **This determination will be complete by November 21, 1996.**

4. **Q:** What was the cause and corrective action(s) taken for the opening of the Transverse Incore Probe (TIP) Ball valves in the presence of a valid Group II containment isolation signal?

A: **Cause:** The design of the TIP control logic is such that while the TIP probe is inserted past the negative 1 inch position (relative to the detector shield), the TIP position encoder senses TIP probe position and signals the computer to open the TIP ball valves. In the event a primary containment isolation signal is received while the TIP probe is inserted, the TIP control system computer directs the TIP drive mechanism to retract the TIP probe. When the probe has been retracted to the negative 1 inch position, the TIP computer directs the TIP ball valves to close, isolating the containment penetration. In the event the TIP probe cannot be retracted, manually initiated explosive shear valves are provided for containment isolation.

During the event, a residual transfer of power to the reserve station service transformers occurred. This resulted in a momentary reduction in voltage to the TIP control system, including the computer.

During the voltage reduction, a 24VDC power supply, which provides power to the control logic for the TIP ball valves, failed. This resulted in a TIP probe position indication of 0 inches because the computer interprets a zero voltage input to correspond to a TIP probe position indication of 0 inches. The computer then opened the TIP ball valves based on a TIP probe position indication of 0 inches.

Corrective Action: The TIP ball valves were declared inoperable (containment isolation) and the containment penetrations isolated in accordance with Technical Specification 3.7.D.2.b. (reference 4). These valves are opened under administrative controls to take TIP traces in accordance with JAF Technical Specification 3.7.D.2.b. (reference 4). The adequacy and basis for the TIP control system is undergoing engineering review. Originally this review was to be complete by February 1, 1997 (JAF commitment No. JAFP-96-0411-04 from LER-96-010, reference 1). This action will be complete prior to plant restart from refuel outage 12 currently scheduled for December 6, 1996.

Change in Commitment Date: JAF commitment No. JAFP-96-0411-04; The adequacy and basis for the TIP control system that is undergoing engineering review will be complete prior to plant restart from refuel outage 12 currently scheduled for December 6, 1996.

5. **Q:** What was the cause and corrective action(s) taken for plant operators switching the RPS bus to a de-energized power source?

A: **Cause:** The control room operating crew mistakenly diagnosed the lack of indication on the control room full core display as an indication that RPS had been de-energized during the transient. In response to this perceived condition, the Shift Manager directed a licensed operator to transfer RPS to its alternate power supplies in order to re-energize RPS. In preparing to transfer, the assigned operator misdiagnosed the power available indications for the RPS power supplies. By transferring RPS to its de-energized alternate power supplies, he de-energized the RPS system.

Possibly contributing to this error was the fact that the method of introducing a residual transfer in the simulator usually resulted in a loss of the RPS M-G Sets. As a result, an initiation technique that more closely emulates actual plant performance has been instituted in the simulator. Other training scenarios are being similarly enhanced.

Corrective Actions:

Prior to plant restart a review was conducted to identify any instances where simulator operator overrides are required to be utilized as simulator malfunctions.

Prior to plant restart operators were trained in the following areas:

- The power supply configuration for the full core display.

- Stress that simulator scenarios closely approximate plant response to abnormal operational conditions but that not all potential equipment failure characteristics can be anticipated or precisely modeled. This training included a discussion regarding the simulator response as it related to the event.

- Stress the principle that operator actions must be deliberate and methodical, particularly when facing uncertain conditions.

- Stress the philosophy that a questioning attitude must be maintained at all times, whether following the steps of a written instruction or the orders of a supervisor.

A follow up memorandum was added to the operator night order book explaining that certain input/output overrides are utilized as simulator malfunctions by grouping them together into common executable instruction sets called Automated Plant Procedures (APPs). The purpose of this memorandum is to remind operators that the simulator response may be slightly different than the plant response for certain APPs. Operators are required to review this information as part of shift turnover in accordance with department standing orders.

Commitment: Simulator scenarios have been reviewed for instances where combinations of input/output overrides, Automated Plant Procedures (APPs), were used in place of simulated malfunctions and may have negatively influenced operator response. New simulated malfunctions have been developed but require testing prior to use in place of the four identified APPs. The identified APPs have been deleted for training use until they are revised. **The testing and use of these malfunctions in their respective scenarios will be complete by January 1, 1997.**

Commitment: The post transient review procedure will be revised to include the requirement to run actual plant transient scenarios on the simulator to validate simulator response. **This action will be complete by December 30, 1996.**

6. **Q:** How was the risk associated with the relay calibration assessed during planning of the work package?

A: The risk assessment and management process for the evolution which initiated this event has three components. Each will be addressed separately

a. **Planning the work package:**

In preparing the work package, an adequate walkdown was not performed. The planner looked at the relay panel, but did not open the back of the panel to inspect the work environment associated with removal and reconnection of the relay capacitor. The individual who planned this work package did not properly evaluate the risk associated with this task.

b. **Removing work from Outage Scope:**

The decision to perform this work at power was predicated on the understanding that the relay being calibrated was not part of a trip circuit during calibration. The proximity of energized equipment with the potential to cause a plant trip during reinstallation of this relay was not recognized.

c. **Preparing to perform the work/performing the work:**

The personnel performing this evolution recognized the risk inherent to the plant but did not question the adequacy of the work plan.

Corrective Action: The procedures which govern the work control, work package planning and the rolling work schedule have been revised (as applicable) to strengthen the process for identifying, communicating and controlling plant work with the potential to have an impact on plant operations. The procedure changes were implemented and training conducted prior to plant restart from the event.

7. Q: Was the IPE utilized in evaluating the risk associated with performing this evolution given the plant configuration (UPS M-G Set out of service) at the time of the event?

A: **Yes**; The JAF work control center did request that the NYPA Probabilistic Risk Assessment (PRA) group evaluate the risk associated with removing the UPS M-G Set from service for corrective maintenance. The PRA group concluded that the additional risk due to having the UPS M-G set out of service for a period of a week was insignificant from the perspective of core melt frequency. The combination of events (UPS M-G set out of service concurrent with the relay work) was not explicitly modeled by the PRA group. The analysis does account for discrete tasks by establishing bounding initiating events which reflect loss of function. As such, the assessment performed by the PRA group did implicitly address the risk associated with performing this relay calibration given the plant configuration (UPS M-G Set out of service).

The PRA group made a second recommendation which identified two potential event initiators:

> *"Please be sure that no other work impacting MCC-252 or L-25 will coincide or we may trip"*

Had the relay work been identified as having the potential to initiate a plant trip, the evolution would have been recognized as a potential event initiator.

After the event, a quantitative analysis of the Core Damage Frequency (CDF) with the UPS M-G set out of service resulted in a predicted CDF of 5E-6. The duration of the plant configuration yielding this CDF was to be one week which yields a conditional Core Damage Probability (CDP) of 5.91E-8. This falls far below the probability cutoff for risk significant temporary changes in the EPRI PSA Applications Guide (reference 9).

This letter contains three commitments, along with a revision to the due date of commitment JAFP-96-0411-04 made in LER-96-010. Attachment I is a summary of those commitments and due dates.

If you have any questions, please contact Mark Abramski of my staff at (315) 349-6305.

Very truly yours,

MICHAEL J. COLOMB
Plant Manager

MJC:MA:las
Attachment

cc: USNRC, Region I
USNRC Resident Inspector
INPO Records Center

Attachment I to JAFP-96-0429

Summary of Commitments

Commitment #	Description	Due Date
JAFP-96-0429-01	The switch evaluated in the reference 5 EFE has been returned to the switch manufacturer for further analysis. This analysis will determine if the switch failure was due to a design or manufacturing condition and will be complete by November 14, 1996. This information will be used to determine if the condition could constitute a "substantial safety hazard" as defined in 10CFR21. This determination will be complete by November 21, 1996.	11/21/96
JAFP-96-0429-02	The post transient review procedure will be revised to include the requirement to run actual plant transient scenarios on the simulator to validate simulator response.	12/30/96
JAFP-96-0429-03	Simulator scenarios have been reviewed for instances where combinations of input/output overrides, Automated Plant Procedures (APPs), were used in place of simulated malfunctions and may have negatively influenced operator response. New simulated malfunctions have been developed but require testing prior to use in place of the four identified APPs. The testing and use of these malfunctions in their respective scenarios will be complete by January 1, 1997. The identified APPs have been deleted for training use until they are revised.	1/1/97
JAFP-96-0411-04	The adequacy and basis for the TIP control system that is undergoing engineering review will be complete prior to plant restart from refuel outage 12 currently scheduled for December 6, 1996.	*12/06/96

*** This date has been revised from the original date submitted under LER-96-010.**