U.S. NUCLEAR REGULATORY COMMISSION

Region I

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Facility Name:	James A. FitzPatrick Nuclear Power Plant
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EXECUTIVE SUMMARY

James A. FitzPatrick Nuclear Power Plant
NRC Inspection Report 50-333/97-08

Operations

- The shutdown for the forced outage conducted on December 7 was safe and well controlled. Good command and control, communication and procedure adherence were noted. Operator observations, involving a degraded residual heat removal system pipe support and mislabeled containment isolation valve, demonstrated good operation practices. The reactor startup following the outage was performed in a safe and prudent manner.

- An operator error was made while performing an electrical ground isolation abnormal operating procedure. Specifically, breakers were operated out of sequence, resulting in the inadvertent automatic operation of high pressure coolant injection (HPCI) system valves. Although the valve operation had minor safety consequences as the HPCI system was out of service for maintenance, the improper performance of an abnormal operating procedure was determined to be a violation. Additionally, the pre-evolution brief for the operations staff was weak in that the assignment of personnel to conduct breaker manipulations was not made.

- The inspector observed portions of the Safety Review Committee meeting conducted on November 20-21, 1997 and noted that the meeting demonstrated good safety oversight of station activities.

Maintenance

- During emergency diesel generator maintenance activities, extensive supervisor involvement was noted. Additionally, pre-evolution briefs were conducted for activities where warranted and procedures were in use. Emergent issues including a lost lube oil valve disc retaining nut and damaged piston assembly resulted in the work activity taking longer than originally scheduled. These emergent issues were effectively addressed through good coordination between operations, maintenance, quality assurance, technical services and supervisor oversight.

- The process to control work activities associated with troubleshooting to locate a direct current ground was unsatisfactory and resulted in an invalid engineered safeguards feature (ESF) actuation signal for the high pressure coolant injection (HPCI) steam supply valves. The HPCI system was out of service for scheduled maintenance. Operators did not recognize that the troubleshooting activities made the primary containment isolation system (PCIS) function inoperable and therefore did not enter the appropriate Technical Specification Limiting Condition for Operation (LCO) action statement. The licensee's immediate corrective actions were appropriate and the root cause analysis was critical of the operation staff's handling of the troubleshooting activities, but lacked in-depth review of the work control process for the activity. Additionally, the licensee's use of junction boxes for temporary storage of parts was considered to be a poor work practice. The failure to enter the TS LCO was a violation.

- The work package to prepare for replacement of the low pressure coolant injection (LPCI) battery was weak in that the impact of removing a portion of the battery enclosure on LPCI battery operability was not considered prior to beginning the work. Although the work was stopped, the licensee subsequently determined that the work would not impact battery operability. Additionally, plant drawings for the structure were not reviewed prior to the work being performed which contributed to confusion in performing the task.

Engineering

- Environmental qualification (EQ) components for the high pressure coolant injection (HPCI) system were erroneously removed from the EQ program in 1993, and in fact, may not have originally met EQ criteria because of installed unrecognized test jacks which affected the EQ of the system. The licensee prepared a justification for continued operation (JCO) which provided reasonable assurance that the equipment would perform its safety function. The licensee was slow to pursue the JCO because the impact of this non-EQ component on HPCI system operability was not initially recognized. Once the problem was recognized, the licensee was aggressive in resolving the issue. The EQ issue was appropriately resolved through removing the component connection to the test jacks and inserting the previously removed components back into the scope of the EQ program. The licensee's erroneous removal of HPCI components from the EQ program was a violation of 10 CFR 50.49.

- The licensee's program to monitor safety relief valve (SRV) leakage was effective. Licensee management exercised good judgement in electing to shutdown the plant to effect repairs to leaking SRVs.

Plant Support

- Overall, the solid radioactive waste program and activities and the program for the transportation of radioactive materials and its related activities were well managed and effective. The quality assurance audits and surveillance reports were thorough, programmatic and well documented.

- Training for personnel involved with solid radioactive waste activities was appropriate in scope and depth. However, the training program was not well organized and documented and therefore the administration of the training program was a weakness.

- Radiological controls in administrative areas relative to the elevated radiation levels due to hydrogen injection were proper and adequate.

- On December 11, 1997, an emergency plan joint drill was conducted with the licensee and Nine Mile Point participating. The emergency preparedness (EP) drill demonstrated solid performance of the EP staff and licensee organization.

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ATTACHMENT

Report Details

Summary of Plant Status

The unit began this inspection period at 100 percent power. On December 7, the plant was taken to cold shutdown to repair leaking safety relief valves (SRVs). The plant was taken critical on December 13 and returned to 100 percent power on December 17. The plant continued operation at 100 percent through the end of the inspection period.

I. OPERATIONS

O1 **Conduct of Operations**[1]

O1.1 Operational Safety Verification

a. Inspection Scope

The inspectors observed plant operation and verified that the facility was operated safely and in accordance with procedures and regulatory requirements. Regular tours were conducted of the plant with focus on safety related structures and systems, operations, radiological controls and security. Additionally, the operability of engineered safety features, other safety related systems and on-site and off-site power sources was verified. The inspectors performed walk downs of accessible portions of several safety related systems.

The inspection activities during this report period included inspection during normal, back shift and weekend hours. Regular tours were conducted of the following plant areas:

control room
secondary containment building
radiological control point
electrical switchgear rooms
emergency core cooling system pump rooms
security access point
protected area fence
intake structure
diesel generator rooms

Control room instruments and plant computer indications were observed for correlation between channels and for conformance with technical specification (TS) requirements. The inspectors observed various alarm conditions and confirmed that operator response was in accordance with plant operating procedures. Compliance with TSs and implementation of appropriate action statements for equipment out of service was inspected. Plant radiation monitoring

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[1]Topical headings such as O1, M8, etc., are used in accordance with the NRC standardized reactor inspection report outline. Individual reports are not expected to address all outline topics.

system indications reviewed for unexpected changes. Logs and records were reviewed to determine if entries were accurate and identified equipment status or deficiencies. These records included operating logs, turnover sheets, system safety tags, and temporary modifications. Control room and shift manning were compared to regulatory requirements and portions of shift turnovers were observed. Daily supervisor meetings were attended to assess personnel focus on risk significant items and plant priorities.

b. Observations and Findings

Overall, the licensee operated the plant safely. Plant activities were performed in accordance with procedures and effective controls were implemented for safe plant operation. Overall, equipment operability, material condition and housekeeping conditions were good.

c. Conclusions

Overall, the licensee operated the plant safely and activities were performed in conformance with requirements. Effective controls were implemented to achieve safe operation of the plant.

O1.2 Plant Shutdown due to Safety Relief Valve Leakage

a. Inspection Scope

During the inspection period the licensee noted an increasing trend in safety relief valve (SRV) leakage to the torus. On December 1, the licensee elected to enter a forced outage in order to replace the leaking SRVs. The inspectors witnessed various portions of the shutdown preparations, power reduction, and reactor cooldown and depressurization activities on December 7. The inspectors' objective was to determine the effectiveness of management controls in ensuring a safe transition to shutdown. In addition, the inspectors observed portions of the reactor startup conducted on December 13, 1997. Inspector attention was focused on reactivity control, operator procedure use and communications.

b. Observations and Findings

The unit was shutdown per operating procedure (OP)-65, Start-up and Shutdown Procedure. Power reduction was performed in accordance with reactor analyst procedure (RAP)-7.3.16, Plant Power Changes, and the main generator was removed from service on December 7, in accordance with applicable operating procedures. The unit was in cold shutdown at 3:48 a.m. and the reactor mode switch was taken to the refuel position at 4:47 a.m. on December 8.

The inspectors noted good command and control of unit shutdown activities. Communications were professional and precise with three-point communications used. Coordination of various shutdown activities by licensed operators was very good. Appropriate oversight of personnel during manipulation of the reactor controls was noted. For example, a second checker for control rod motion and selection was stationed. In addition, senior licensee management personnel were assigned for shift coverage.

Prior to the startup of the plant, the operations staff noted two plant deficiencies. One deficiency involved dislodged grouting material from behind a pipe support in the residual heat removal system, and the second issue involved a mislabeled containment isolation valve. Both issues were adequately addressed by the licensee prior to startup, and reviewed by the inspectors. The latter event will be further reviewed following the issuance of a licensee event report (LER).

The startup was characterized by clear operator communications and procedure use, attentive management oversight, and effective control by shift supervision. Shift turnover meetings were performed in a controlled manner and crew briefings were good. Senior operations management personnel were designated to provide continuous oversight.

c. <u>Conclusions</u>

The shutdown for the forced outage was safe and well controlled. Good command and control, communication and procedure adherence were noted. The observations by the operators demonstrated good operational practices. The reactor startup following the outage was performed in a safe and prudent manner.

O4 Operator Knowledge and Performance

O4.1 Battery Ground Isolation Procedure Error (Violation 50-333/97007-01)

a. <u>Inspection Scope</u>

On October 23, the operators entered abnormal operating procedure (AOP)-23, Direct Current (DC) Power System Ground Isolation, in response to indications of a ground on the "B" battery. Testing involving the high pressure coolant injection (HPCI) logic system had just been completed prior to the ground appearing on the control room instrumentation, so the control room staff elected to proceed to the portion of the AOP which isolates the HPCI logic circuitry. During performance of the procedure, the operators failed to open the power supply breakers for 23MOV-57 and 23MOV-58, the HPCI booster pump suction from the suppression pool downstream and upstream isolation valves respectively. This resulted in the valves automatically opening when the correct circuit breaker, 71DCB2 Breaker 6, HPCI Logic Power Supply, was opened in the improper sequence. The event occurred during a HPCI maintenance limiting condition for operation (LCO) and therefore the system was already considered inoperable. The inspector reviewed procedures, plant logs and conducted interviews with station personnel involved in the performance of the ground isolation procedure.

b. Findings and Observations

AOP-23, DC power System B Ground Isolation, provides steps which attempt to locate the source of a ground in the DC power system. The procedure contains general steps in the main body of the text and lists the specific breakers to be utilized in the isolation of the grounds in an attachment to the procedure. In brief, the procedure directs the operators to establish communications between the control room and the operator at the specified breaker, perform any actions required by the breaker attachment sheet, enter the any applicable LCOs, and open the isolation breaker. The ground detector in the control room is then monitored to see the effect, if any, of opening the isolation breaker. The process is repeated until the ground is isolated. More specifically, in the attachment to the procedure, tables identify the isolation breaker to be opened, its corresponding circuit or component, and the actions required prior to opening the isolation breaker. In this particular event, the operator became focused on selecting the proper isolation breaker and omitted the requirements of the procedure to open the power supply breakers for 10MOV-57 and 10MOV-58.

In discussion with the plant staff, the inspector learned that the pre-evolution brief for the operations staff was not specific. The operators had been monitoring the ground circuit prior to the alarming condition being reached, taken out the AOP, and discussed the most probable circuit to check based on recent HPCI system testing. The inspector noted that all the control room staff had been included in the discussions of current plant conditions, including the selection of an additional operator to perform a peer check of the isolation breaker operation. However, the assignment or discussion of who was going to open the breakers to 10MOV-57 and 10MOV-58 was not discussed as part of the brief.

The impact of the procedure error was to cause the HPCI booster pump suction to shift from the condensate storage tank (CST) to the torus. The torus suction valves are designed to go open on low CST water level or high suppression pool level to ensure that HPCI has a makeup water source. This action occurred because the HPCI logic circuitry, following a power loss to the CST level instrumentation when breaker six was opened, caused the suction valves to automatically go open. As previously stated, the system was undergoing maintenance and thus was already considered inoperable. The impact was limited to unnecessarily challenging the HPCI logic circuitry and cycling valves. Immediate correction actions were to restore the power to the logic circuitry, reposition the valves, and re-perform the procedure correctly. The electrical ground was subsequently located and fixed.

c. Conclusions

The operator error in performing the actions of the AOP had minor safety consequences, however, the proper performance of abnormal operating procedures is of high importance and was determined to be a violation. **(50-333/97008-01)**

Additionally, the pre-evolution brief for the operations staff was weak in that the assignment of personnel to conduct breaker manipulations was not made.

O7 Quality Assurance in Operations

O7.1 Licensee Self-Assessment Activities

a. Inspection Scope

During the inspection period, the inspectors reviewed multiple licensee self-assessment activities, including portions of the Safety Review Committee (SRC) meeting conducted on November 20 - 21, 1997. Observations of the SRC meeting are noted below.

b. Observations and Findings

Recent plant history and issues and performance indicators, operational review and human performance trends were discussed. Specific issues that were discussed in depth included nuclear personnel turnover and engineering lack of rigor. SRC members demonstrated a good questioning attitude and good interaction with the Indian Point 3 representative were noted. Follow-up items were developed where appropriate.

c. Conclusions

The SRC meeting demonstrated good safety oversight of station activities.

II. MAINTENANCE

M1 Conduct of Maintenance

M1.1 General Comments on Maintenance and Surveillance Activities

a. Inspection Scope

The inspectors observed selected maintenance activities to verify that activities were conducted in a manner sufficient to ensure reliable, safe operation of the plant. The inspectors observed selected surveillance tests to determine whether the tests were conducted in accordance with technical specification and other requirements.

The inspectors observed all or portions of the following work activities:

WR 97-06988-06	Replace "B" control rod drive (CRD) pump and restore temporary modification 97-095.
WR 97-08389-01	Investigate and repair source of water leakage in main stack room.

WR 97-08063-02	Troubleshoot/repair valve positioner, feedwater heater 33E-5A drain valve operator.
WR 97-06476-00	Troubleshoot/repair pre-cooler drain lines.

The inspectors observed portions of the following surveillance activities:

ST-3P	Core spray flow rate and valve inservice test.
ST-5D	Average Power Range Monitor Calibration.
ISP-94	Reactor Protection System Electrical Protection Assembly Functional Test/Calibration.

b. Observations and Findings

The inspectors found the work performed under these activities to be professional and thorough. Technicians were experienced and knowledgeable of their assigned task. Activities were conducted appropriately and in accordance with procedural and administrative requirements. Good coordination and communication were observed during performance of the surveillance activities.

c. Conclusions

Overall, the above maintenance and surveillance activities were well conducted, with good adherence to both administrative requirements and maintenance and surveillance procedures.

M1.2 "A" Emergency Diesel Generator Scheduled Maintenance

a. Inspection Scope

The "A" emergency diesel generator (EDG) was scheduled for planned maintenance from November 3 to 5, 1997. Activities to be completed included routine preventive and corrective maintenance, fuel oil replacement, and power pack assembly (cylinder liner, pistons and associated components) replacement. The inspector observed selected activities including procedure use, quality assurance, and supervisor oversight. During the planned maintenance, emergent work including a replaced power pack failure and the identification of a missing nut on a lube oil check valve were also reviewed.

b. Observations and Findings

During performance of maintenance procedure (MP) 93.11, the lube oil gallery supply check valve was inspected due to industry information which documented a history of problems with the valve. Mechanics identified that the valve disc retaining nut was missing, and the disc was lying on the bottom of the valve. The valve is a ½ inch swing check valve. The licensee initiated deficiency and event report (DER) 97-1545 to investigate the problem and to analyze the impact of the missing nut. The check valve is located between the lube oil cooler and main lube oil pump discharge in a line used for lube oil warm up when the engine is shutdown. On November 5, a licensee quality assurance (QA) inspector found a brass nut of similar proportions located outside the screenwell and the licensee determined that the nut was the missing nut.

The licensee's basis for this determination was that the lube oil cooler had been disassembled prior to the valve inspection. The fact that the nut was missing was not known at the time of the lube oil cooler inspection. Detection of the nut during lube oiler cooler maintenance would be difficult, due to the size of the nut and the amount of oil present. Since the nut was not found in the locations where it would be expected to be based on lube oil flow paths, the licensee concluded that the nut was removed from the cooler without detection during cooler maintenance. The inspector concluded that the licensee's analysis was reasonable.

Another major activity completed was that the power pack assemblies for all cylinders were replaced. During EDG post work testing, a high crankcase pressure alarm was observed after about 15 minutes of EDG operation and the engine was shutdown. It was determined that one piston and liner was damaged. Specifically, the bottom end of the piston skirt was broken and other internal parts were damaged. The assembly was removed and replaced with a rebuilt assembly, broken parts were retrieved from the lube oil sump and the lube oil strainer and filter were changed. The licensee is awaiting the results of an equipment failure evaluation for the damaged power pack assembly.

The EDG limiting condition for operation (LCO) was exited on November 8. The delay in completing the work activities was a result of emergent work.

c. Conclusions

Extensive supervisor involvement was noted. Additionally, pre-evolution briefs were conducted for activities where warranted and procedures were in use. Emergent issues including the lost lube oil valve disc retaining nut and the damaged piston assembly resulted in the work activity taking longer than originally scheduled. These emergent issues were effectively addressed through good coordination between operations, maintenance, quality assurance, technical services and supervisor oversight.

M4 Maintenance Staff Knowledge and Performance

M4.1 Invalid Engineered Safeguards Feature (ESF) Actuation and Failure to Perform Technical Specification Required Actions While Performing Troubleshooting (Violation 50-333/97008-02)

a. Inspection Scope

On October 24, while performing troubleshooting to locate a ground on the "B" DC power system, an inadvertent short across a pair of test jacks in an electrical panel caused a partial isolation signal for the high pressure coolant injection (HPCI) system. The HPCI system was not operating at the time and was in a scheduled limiting condition for operation (LCO) for maintenance. The expected isolation of the appropriate HPCI steam supply valves did not occur. The operations staff subsequently determined that fuses removed from the HPCI logic circuitry during troubleshooting, prevented the associated isolation valves from automatically closing. The maintenance activity disabled the primary containment isolation function of the HPCI valves without entering the appropriate LCO. Following the discovery of this, the licensee completed the actions required by Technical Specifications (TS) by isolating the outboard HPCI steam isolation valve, 23MOV-60. The inspector reviewed the licensee's root cause analysis for the event, conducted interviews, attended management meetings on the event, and reviewed station procedures to assess the event regarding safety significance and work control processes.

b. Observations and Findings

On October 24, 1997, maintenance activities to repair a ground problem were conducted which rendered the primary containment isolation system (PCIS) function of the outboard HPCI steam isolation valves inoperable, however, the applicable LCO action statement was not entered. If one or more of the containment isolation valves are inoperable, Technical Specifications require, in part, that the affected penetration be isolated within four hours by use of at least one deactivated automatic valve secured in the closed position. Operators did not recognize that PCIS was disabled until after a maintenance error caused a short of the logic circuitry which caused an invalid engineered safeguards feature (ESF) actuation signal sixteen hours after disabling the logic.

The root cause analysis identified several inappropriate actions. These included the failure to recognize the impact on the PCIS function of the HPCI isolation valves when removing logic fuses during surveillance test ST-2M, ECCS Trip Systems Bus Power Monitors Functional Test, disabling the same PCIS function during trouble shooting without entering the applicable LCO, and failing to enter the correct LCO when the condition was recognized. Several causes were identified by the licensee for the inappropriate actions identified above. Surveillance test ST-2M was inadequate, in that it did not recognize disabling the PCIS function of the HPCI valves, a less than adequate review of the short form temporary operating procedure and protective tag out, and inadequate training on a previous technical specification change which resulted in operators using the incorrect section of the TS. The licensee developed twelve recommended corrective actions, including revising procedures to capture the lessons learned, training and review of the event with operators, and review of all surveillance test and operating procedures to identify the impact of fuse removal on TS.

The inspector also reviewed the troubleshooting work request which led to the event and determined that the impact on PCIS was also missed during the work control process. The inspector noted that this issue was not addressed in the root cause analysis. The licensee subsequently reviewed the work control process and determined that the troubleshooting process for this emergent work item relied on the workers to determine the affects of their actions in the field during work execution. Had a detailed review of the logic by the workers and the operations staff been conducted, the potential impact of the work could have been identified. The mind set of the plant staff was that the system was in an existing LCO and tagged out, therefore work would not impact the plant. This was an incorrect assumption as identified when a technician inadvertently shorted two terminals in a junction box. The issues surrounding the performance of the troubleshooting were discussed in Licensee Event Report 97-011 and will be addressed when the LER is reviewed.

c. Conclusions

The process to control work activities associated with troubleshooting to locate a DC ground was unsatisfactory and resulted in an invalid ESF actuation. Operators did not recognize that the troubleshooting activities made the PCIS function inoperable and therefore did not enter the appropriate LCO action statement. The licensee's immediate corrective actions were appropriate and the root cause analysis was critical of the operations staff's handling of the trouble shooting activities but lacked in-depth review of the work control process for the activity. The failure to enter the LCO was determined to be a violation **(50-333/97008-02)**.

M4.2 Low Pressure Coolant Injection (LPCI) Battery Replacement

a. Inspection Scope

The inspector observed preparations for the "B" LPCI battery replacement in the reactor building. The mechanics were trying to remove several sections of metal panels that make up one of the walls to "B" LPCI battery enclosure. In discussion with the maintenance personnel the inspector learned that the wall was much more intricate than the maintenance crew had expected. The responsible engineer was notified and after further discussion the licensee determined that the work should not be continued. A horizontal top corner piece of the structure had been removed to allow access to the vertical wall sections, but no other pieces were removed. The inspector reviewed the licensee's work planning and discussed the activity with the licensee personnel.

b. Observations and Findings

Work request (WR) 96-05333-07, was written to remove panels from the west wall of the "B" LPCI enclosure, to facilitate the installation of a temporary load handling monorail. The monorail was to be used to replace the existing LPCI battery cells with new cells during the upcoming scheduled LCO maintenance period. In follow up interviews with the plant staff the inspector learned that the original maintenance package did not consider potential fire protection and seismic issues associated with the removal of various battery enclosure panels. In discussion with the planning staff the inspector discovered that the enclosure drawings were not reviewed as part of the work package planning which contributed to a lack of detail in the work package. The licensee initiated a deficiency event report to investigate the issue and utilized another method to exchange the battery cells. The licensee's investigation concluded that the battery enclosure was not necessary for LPCI battery operability. The battery replacement work package was weak in that the impact of the work activity was not assessed. Additionally, the work control procedure did not include a requirement to include all structures, systems and components (SSCs) when reviewing work for seismic concerns. The corrective actions were appropriate for the above findings.

c. Conclusions

The inspector concluded that the work package to prepare to replace the LPCI battery was weak in that the impact of the work on the LPCI battery operability was not considered prior to beginning the work. Although the work was stopped, the licensee subsequently determined that the work would not impact battery operability. Additionally, plant drawings for the structure were not reviewed prior to the work being performed which contributed to confusion in performing the task.

III. ENGINEERING

E1 Conduct of Engineering

E1.1 Environmental Qualification of components in the High Pressure Coolant Injection System (Violation 50-333/97008-03)

a. Inspection Scope

On October 24, 1997, while performing troubleshooting for a DC ground, a nut was dropped across test jacks located in a junction box. The resulting short caused a HPCI isolation signal. The identification of electrical test jacks on junction boxes for HPCI and RCIC isolation circuits raised questions concerning the operability and environmental qualification (EQ) of the associated components. The inspector reviewed the licensee's EQ program calculations, justification for continued operation (JCO) and conducted a physical walkdown of the affected areas.

b. Observations and Findings

On October 24, 1997, during troubleshooting on a pressure switch for the source of a DC ground, a nut was dropped across two hot test points in a junction box, located in the west crescent area, which initiated a HPCI isolation trip signal Fuses pulled for the troubleshooting prevented the actual system isolation. The inspector noted that the junction box was marked as EQ, however, the pressure switches located in the junction box had been removed from the EQ program. Test jacks were also located in the bottom of three additional junction boxes and were not identified on plant drawings. The concern was that the test jacks may not maintain electrical integrity in a high energy line break (HELB) and therefore the potential existed to impact the HPCI steam line isolation function. Similar test jacks were located in junction boxes associated with RCIC.

At the time of the event, the licensee did not initially recognize the need to determine the operability of the affected components. Subsequently, an operability review for HPCI and RCIC was completed on November 4, 1997, and the licensee prepared a JCO, JAF-EQ-JCO-97-002, Plant Operation with Test Jacks Installed in Junction Boxes JB-R2550D and JB-R2550E for 23 PS-86A,B,C and D to justify continued operation with the test jacks installed. The licensee's operability review determined that the test jacks did not affect the operability of the circuits.

The licensee reviewed the EQ status of the associated junction boxes. It was determined that on March 3, 1993, the licensee deleted approximately 15 components from the EQ program for harsh environment plant electrical equipment. The analysis was documented in JAF-CALC-HPCI-00820 and was prepared to show that HPCI electrical components would not be subject to a harsh environment during a HELB. The licensee determined that a nonconservative assumption was made in the calculation which resulted in removing the HPCI components from the EQ program.

The licensee's corrective actions included walkdowns to identify any other similar test jacks that posed EQ issues. The results of the walkdown determined the extent of the condition was limited to HPCI and RCIC. Additionally, the licensee removed the electrical connections to the HPCI and RCIC test lugs under a plant modification. A longer term action review other components removed from the EQ program was in progress.

The inspector noted a station work practice where technicians occasionally used junction boxes to temporarily store various objects while working on components. Typically, this practice was used when technicians were working on grates where there was not a readily available place to temporarily store small tools or components. The licensee reviewed their practices in this area and determined that this practice would no longer be used.

c. Conclusions

Following the initial event, the licensee was slow to pursue the JCO because the EQ aspects were not readily recognized. The EQ components were erroneously removed from the program in 1993, and in fact, did not originally meet EQ criteria because of the unrecognized installed test jacks. A JCO was prepared which provided reasonable assurance that the equipment would perform its safety function. The EQ issue was appropriately resolved through removing the connection to the test jacks and inserting the previously removed components into the scope of the EQ program. 10 CFR 50.49, Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants, describes EQ program requirements. Contrary to these requirements, the licensee erroneously removed HPCI components from the EQ program **(VIO 50-333/97008-03)**.

E2 Engineering Support of Facilities and Equipment

E2.1 Licensee Monitoring of Leaking Safety Relief Valves (SRVs)

a. Inspection Scope

Because of a history of safety relief valve leakage and an industry event where an SRV inadvertently opened, the licensee monitors SRV leakage. The inspectors reviewed the licensee's SRV monitoring program and discussed the issue with licensee personnel. Total SRV leakage recently increased to a point where the licensee elected to shutdown and repair the leaking SRVs as described in Section O1.2.

b. Observations and Findings

The licensee uses 11 Target Rock 2 stage pilot actuated safety relief valves for pressure relief of the reactor vessel. The licensee monitors SRV tailpipe temperature and calculates leakage based on torus heat up rate. The licensee had previously developed an action plan to schedule a plant shutdown at a torus heatup rate corresponding to an SRV leakage rate of 400 lbs/hr and to shutdown the plant at a torus heatup rate corresponding to 600 lbs/hr. As of November 21, the licensee was operating with indication of 3 leaking SRVs and a leak rate of 450 lbm/hr with most leakage attributed to "C" SRV main seat leakage.

The inspectors monitored the licensee's performance related to SRV leakage. The licensee closely tracked SRV performance through daily torus heat up rate calculations and observations of SRV tailpipe temperature. In addition, the inspectors noted that SRV performance is routinely scheduled for discussion at the department manager's meetings.

c. Conclusions

The licensee's program to monitor SRV leakage was effective. Licensee management exercised good judgement in electing to shutdown the plant on December 7th to effect repairs to leaking SRVs.

E8 Miscellaneous Engineering Issues

E8.1 (Closed) Inspector Follow-up Item (IFI) 50-333/96007-02: Affect of reactor water cleanup and control rod drive flow on alterna[illegible] decay heat removal (ADHR) pre-operational testing. During refueling outage 12, the inspectors noted that the control rod drive (CRD) and reactor water clean-up (RWCU) systems were in service providing approximately 240 gallons per minute flow to the reactor vessel and providing additional refueling cavity mixing during the pre-operational testing of the ADHR system. This was of concern to the inspectors because the intent of the pre-operational testing was to ensure that the alternate decay heat removal system was capable of removing the heat generated by the spent fuel in both the reactor vessel and in the spent fuel pool. The test was to demonstrate that the

ADHR system would remove heat from the reactor cavity and the spent fuel pool using natural circulation. The inspector's concern was that the added circulation and cooling water may result in non-conservative results with respect to the capabilities of the system. Additionally, the original calculations for the heat removal capacity did not account for the additional circulation provided by the CRD and RWCU systems, which were in service during the test. The licensee subsequently revised the computer model used to independently verify the General Electric design calculations on the ADHR system, JAF-CALC-DHR-02380, ADHR System Thermal Hydraulic Analysis, and incorporated the CRD and RWCU system flows. The revised calculation demonstrated that there was no significant change in the natural circulation flow characteristics of the reactor vessel and the surface water temperature of the SFP differed from the original calculation by one degree Fahrenheit, which was considered to not have a material impact.

E8.2 (Closed) Unresolved Item (URI) 50-333/95006-03: Components Missing from Control Room Ventilation Drawings. As part of a system walkdown, the inspector identified that two motor operated dampers (MODs), MOD-113 and MOD-114, and flow element (FE)-102 were omitted from as-built drawing FB-35C, Rev. 12, Equipment Room Heating, Vent and Air Conditioning. However, the inspector noted that the components were identified on the control room flow diagram FB-45A and identified in the Final Safety Analysis Report (FSAR). The inspector was concerned that despite a temporary modification and a minor modification being processed for two safety-related components, this deficiency in the as-built drawing was not identified by the licensee. The licensee subsequently determined that the error occurred during original construction and that the drawing was not required to be updated because of its classification. The System Engineering Standing Order, (SESO)-2, classified the drawing as a type "C" drawing, and per Design Change Manual, (DCM)-22, drawing changes are not required until five changes have been posted to the drawing or with the department manager's approval. The classification of the drawing signifies that it is used to facilitate design and maintenance that has a low frequency use. The current practice is for users to verify the current status of changes to drawings in the drawing status log prior to use. In the review of the event, however, the licensee determined that the modification process should have listed the drawing as an "affected drawing" so that the appropriate revision process would have been implemented. The modification removed the motors from the dampers and put them in a fail safe position and they currently do not provide any safety function. The inspector reviewed the corrective actions including revision of the appropriate drawings, updating the plant equipment database and training on the issue. The inspector determined the corrective actions were appropriate and that a violation did not exist because the licensee practices were in accordance with their procedural requirements for drawing update and were appropriate to the circumstances.

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Implementation of the Solid Radioactive Waste Program

a. Inspection Scope

The inspector reviewed the licensee's solid radioactive waste program. Information was gathered through observation of activities, tours of the radiologically controlled areas including the radioactive waste building, discussions with cognizant personnel, and review and evaluation of procedures and documents.

b. Observations and Findings

In reviewing the implementation of the solid radioactive waste program, the liquid waste processing and solidification methods were inspected, the dry active waste operation was evaluated, and storage locations were inspected. Liquid radioactive waste was processed through a vendor filter/demineralizer skid. There was a "green is clean" program, and potentially contaminated materials were sorted and frisked to minimize the generation of radioactive waste. Offsite contracted services were available for equipment/parts decontamination and for supercompaction or incineration of dry active waste. The interim waste storage building provided five high bay areas for storage of low level waste and a separate area with shielded concrete cells for storage of higher level waste such as high integrity containers (HICs) filled with spent powder or resin. A large portion of this storage capacity was still available.

The volume of solid radioactive waste and especially of dry active waste had steadily and significantly decreased over the last several years. Numerous initiatives to reduce waste were evident. This included the establishment of a low-level radioactive waste reduction team. Reusable wrist and ankle straps in place of masking tape for protective clothing purposes, reusable bags, the wearing of cotton glove liners into the whole body contamination monitors, and the evaluation/implementation of good radioactive waste reduction practices from other licensees and from a utility research group have contributed to the decrease in radioactive waste volume.

Housekeeping was good; aisle ways were clear and clean; storage areas were clean and orderly; contaminated areas were minimized; radioactive material was clearly and properly labeled and stored in an orderly fashion.

c. Conclusions

The implementation of the solid radioactive waste program was well managed and effective.

R1.2 Compliance with NRC and Department of Transportation (DOT) Regulations for Shipping of Low Level Radioactive Waste (LLRW) for Disposal and Transportation of Other Radioactive Materials

a. Inspection Scope

The inspector reviewed the licensee's transportation of radioactive materials. Information was gathered through observation of activities, discussions with cognizant personnel and review and evaluation of procedures and documents. Temporary Instruction (TI) 2515/133, Implementation of Revised 49 CFR Parts 100-179 and 10 CFR 71 was completed during this review.

b. Observations and Findings

The shipping records for several past radioactive waste shipments were reviewed. These records were found to be appropriate and complete The inspector observed the transfer of a HIC containing spent resin into a shipping cask. The waste classification and Department of Transportation (DOT) shipment type determination for this shipment were evaluated and met regulatory requirements. After this waste shipment was on the public highway, the inspector tested the emergency response information telephone process (10 CFR 49, Subpart G, Emergency Response Information) by calling, in the evening, the emergency response telephone number which was on the radioactive waste manifest. The inspector's call was answered, and the emergency response information described in 10 CFR 49.602 was made available by the recipient of the call in a timely manner.

c. Conclusions

Good performance was demonstrated in the area of packaging and transportation of solid radioactive waste.

R1.3 Elevated Radiation Exposure Levels Due Hydrogen Injection

a. Inspection Scope

The inspector reviewed the licensee's radiological controls in administrative areas relative to the elevated radiation levels due to hydrogen injection into the reactor coolant system. Information was gathered through observation of a radiation survey, conduct of a radiation survey, tours of the affected locations, discussions with cognizant personnel, and review and evaluation of procedures and documents.

b. Observations and Findings

Elevated dose rates outside the radiologically controlled area (RCA) due to hydrogen injection, and the radiological controls in those areas were reviewed. This review focused on the second floor of the old administration building and on the second floor of warehouse No. 1 since these areas were representative of the affected areas which were outside the RCA, within the protected area, and which were occupied by mainly administrative personnel. These locations were included in the licensee's routine radiation survey program.

Dose rates on elevation 290 of warehouse No. 1 varied from 10 to 70 microrem per hour with reactor power level at 100% based on the licensee's routine radiation survey results. Dose rates on elevation 286 of the old administrative building varied from 10 to 150 microrem per hour with reactor power level at 100% based on the licensee's routine radiation survey results and on an independent survey by the inspector using a calibrated licensee radiation survey meter. At the time of these surveys, the hydrogen injection rate was approximately 18.5 cubic feet per minute. The highest dose rates in the latter area were along the outside window areas, and the dose rates gradually decreased with distance away from these window areas.

Current occupancy factors were observed for each of the two areas and appeared to approximate 40 hours per week. Assuming a 40-hour work week, continuous occupancy during the work week, and 50 work weeks per year, 150 and 10 microrem per hour would equate to 300 and 20 millirem per year, respectively. For perspective, the average background radiation level in the United States is about 10 microrem per hour or approximately 100 millirem per year (24 hours per day and 365 days per year).

10 CFR 19.12, "Instruction to Workers," requires that all individuals who in the course of their employment are likely to receive in a year an occupational dose in excess of 100 millirem shall receive radiation protection instruction commensurate with potential radiological health protection problems present in the workplace. Radiation safety training records for several white-badged individuals (non-radiation workers) in each of these areas were inspected. The individuals were confirmed to have received general employee training, which includes basic radiation protection training on an annual basis. The inspector confirmed that the training included discussion of the guidance in Regulatory Guide 8.13, "Instruction Concerning Prenatal Radiation Exposure."

10 CFR 20.1502 requires the use of individual radiation monitoring devices for adults likely to receive a dose in excess of a total effective dose equivalent of 500 millirem per year; and for the embryo/fetus of a declared pregnant women for whom the embryo/fetus is likely to receive a dose in excess of 50 millirem during the entire pregnancy. Inspector observations noted that a number of the administrative (non-radiation worker) personnel had been provided individual radiation monitoring devices even though this was not required by 10 CFR 20.1502. In some of these administrative areas, one would be likely to receive a dose in excess of 50 millirem within a nine month period, and, in such a case, 10 CFR 1502 would require the use of a individual radiation monitoring device for the embryo/fetus of a declared pregnant woman. The inspector confirmed that a declared pregnant woman had been provided such dosimetry.

The inspector also reviewed licensee documents titled "Radiological Technical Information Document (RTID) No. 93-011, Basis for Individual Monitoring Requirements Within the Restricted Area of the JAFNPP, December 13, 1993," "First Quarter 1997 Restricted Area Dose Evaluation, April 23, 1997," "Second Quarter 1997 Restricted Area Dose Evaluation, July 18, 1997," "Third Quarter 1997 Restricted Area Dose Evaluation, November 3, 1997," and "RTID-96-005, Radiological Assessment of Doses/Dose Rates in Non RCA Occupied Areas, June 19, 1996."

Based on monitoring and survey data by the licensee and the NRC, observations in several affected areas, and review of training and documents describing the licensee's evaluation of this issue, the inspector concluded that the radiation badging requirements in 10 CFR 20.1502 and the training requirements in 10 CFR 19.12 were being followed.

c. Conclusions

Radiological controls in administrative areas relative to the elevated radiation levels due to hydrogen injection were proper and adequate.

R5 Staff Training and Qualification in RP&C (Inspector Follow-up Item (IFI) 50-333/97-008-04)

a. Inspection Scope

The inspector reviewed the qualifications and training of selected radioactive waste personnel. Information was gathered through discussions with cognizant personnel, and review and evaluation of documents.

b. Observations and Findings

Training department personnel stated that the training for radioactive waste processing, handling/transferring, packaging, and shipping was provided by contractors and that the training courses were reviewed and approved by licensee personnel before implementation. The inspector reviewed the course materials used for the training of the radioactive waste handlers and shippers. The scope and depth of the course materials was fully adequate. The inspector verified that these individuals had been recently trained in the aforementioned topics and that the two individuals who were responsible for classifying waste and determining DOT shipment type had been retrained on the applicable computer program in mid 1997. Additionally, it was confirmed that all individuals authorized to sign shipping paperwork had received recent training on the shipping regulations.

However, a documented description of the required training for radioactive waste processors, handlers/transferors, classifiers, and shippers was not available, and the training record database was incomplete in that the latest training for the computer program used for classifying and typing waste shipments had not been entered. These administrative deficiencies were considered a program weakness.

The licensee stated that a matrix of required training and the frequency of same for radioactive waste processors, handlers/transferors, classifiers, and shippers would be developed and kept available for review and that the training record database would be updated and kept updated. This issue will be reviewed during a subsequent inspection **(IFI 50-333/97008-04)**.

c. Conclusions

The training and retraining for the radioactive waste handlers/transferors and shippers was appropriate in scope and depth, and records of this training were adequately maintained. However, a documented description of required training was not available, and the training record database was incomplete. Therefore, the training program was not well organized and documented.

R7 Quality Assurance in RP&C Activities

a. Inspection Scope

The inspector reviewed the licensee's quality assurance (QA) activities for solid radioactive waste management and transportation of radioactive materials. Information was gathered through discussions with cognizant personnel and review and evaluation of documents.

b. Observations and Findings

Audit A96-17J, conducted in the Fall of 1996, covered the implementation of the revised DOT and NRC radioactive material shipping requirements. This audit resulted in two Deviation and Event Reports (DERs) and two Recommendations. The inspector's review of the audit checklist and audit report showed that the audit was thorough and programmatic.

Audit A97-05J, conducted in February of 1997, covered the Process Control Program (PCP) and Regulatory Guide 1.21. This audit resulted in the issuance of two DERs and one Recommendation in the PCP area. The audit checklist and audit report portions dealing with the PCP were reviewed and were found to be in-depth efforts.

Six QA surveillance reports, performed from November 1996 to September 1997, were evaluated. These reports covered receipt inspections of radioactive waste shipping casks and liners, shipment inspections of radioactive waste in casks, review of documentation packages for several waste shipments, and the release of material from the radiologically controlled area. There were no resultant DERs or recommendations based on these reports. The surveillance reports showed that the surveillance activities were detailed and well documented.

c. Conclusions

The Quality Assurance audits and surveillance reports were thorough, programmatic, and well documented.

R8 Miscellaneous RP&C Issues

R8.1 (Closed) Violation 50-333/96007-08: Failure to follow plant Technical Specification for locked high radiation area entry (i.e., contractor in the drywell with his alarming dosimeter turned off). The inspector reviewed the corrective actions described in the licensee's response letter dated February 21, 1997. The corrective actions were reasonable and comprehensive. No similar problems were identified.

R8.2 (Closed) Violation 50-333/96007-09: Failure to follow a formal quality assurance program (i.e., failure to promptly identify and correct a deviation involving the lack of current certification of technicians for use of a computer program code used to classify shipments). The inspector reviewed the corrective actions described in the licensee's response letter dated February 21, 1997. The corrective actions were appropriate and complete. No similar problems were identified.

P1 Conduct of EP Activities

P1.1 Emergency Plan Drill

a. Inspection Scope

On December 11, 1997, an emergency plan joint drill was conducted with the licensee and Nine Mile Point participating. The purpose of the drill was to demonstrate that various emergency preparedness functions could be performed jointly from the emergency operations facility (EOF). The drill was a partial scale drill and had limited participation by Oswego County.

The inspector observed and evaluated the performance of licensee emergency response personnel in the EOF including staffing and activation; facility management and control; accident assessment and classification; offsite dose assessment; protective action decision making and implementation; notifications and communications; and interaction with the Oswego County personnel.

b. Observations and Findings

The emergency was properly classified. The reactor condition and emergency was continuously reassessed. Environmental sampling teams were appropriately deployed. Offsite dose assessment and protective action recommendations were appropriate. Communications within the Emergency Operations Facility were frequent with proper notifications and interaction with county personnel noted. A particular strength noted was the good coordination between the emergency directors from the Nine Mile Point and FitzPatrick facilities in setting priorities.

c. Conclusions

The emergency preparedness drill demonstrated solid performance of the EP staff and licensee organization.

P8 **Miscellaneous EP Issues (EA 98-008) (NCV 97-008-05)**

(Closed) IFI 50-333/97002-03(NCV 97-008-05): Adequacy of emergency procedures governing evacuation from areas near the new fuel storage vault and failure to meet requirements of 10 CFR 70.24 for new fuel criticality monitors. This issue involved the failure to have in place either an adequate criticality monitoring system for storage and handling of new (non-irradiated) fuel or an NRC approved exemption to this requirement contained in 10 CFR 70.24. The issue was previously left as an inspector follow-up item pending additional internal NRC guidance regarding the adequacy of the existing monitoring system as well as the emergency procedures governing evacuation from areas near the new fuel storage vault.

10 CFR 70.24 requires that each licensee authorized to possess more than a small amount of special nuclear material (SNM) maintain in each area in which such material is handled, used or stored a criticality monitoring system which will energize clearly audible alarm signals of accidental criticality occurs. The purpose of 10 CFR 70.24 is to ensure that, if a criticality were to occur during the handling of SNM, personnel would be alerted to that fact and would take appropriate action.

Most nuclear power plant licensees were granted exemptions from 10 CFR 70.24 during the construction of their plants as part of the Part 70 license issued to permit the receipt of the initial core. Generally, these exemptions were not explicitly renewed when the Part 50 operating license was issued, which contained the combined Part 50 and Part 70 authority. In August 1981, the Tennessee Valley Authority (TVA), in the course of reviewing the operating licenses for its Browns Ferry facilities, noted that the exemption to 10CFR 70.24 that had been granted during the construction phase had not been explicitly granted in the operating license. By letters dated August 11, 1981, and August 31, 1987, TVA requested an exemption from 10 CFR 70.24. On May 11, 1988, NRC informed TVA that "the previously issued exemptions are still in effect even though the specific provisions of the Part 70 licenses were not incorporated into the Part 50 license." Notwithstanding the correspondence with TVA, the NRC has determined that, in cases where a licensee received the exception as part of the Part 70 licenses issued during the construction phase, both the Part 70 and Part 50 licenses would be examined to determine the status of the exemption. The NRC view now is that unless a licensee's licensing basis specified otherwise, an exemption expires with the expiration of the Part 70 license. The NRC intends to amend 10 CFR 70.24 to provide for administrative controls in lieu of criticality monitors.

The NRC has concluded that a violation of 10CFR 70.24 existed at FitzPatrick due to the inadequacy of the existing new fuel vault radiation monitor as a comprehensive criticality monitoring system for new fuel handling and storage. The NRC has also determined that numerous other licensees have similar circumstances that were caused by confusion regarding the continuation of an exemption to 10 CFR 70.24 originally issued prior to issuance of the Part 50 license. After considering all the factors that resulted in these violations, the NRC has concluded that while a violation did exist, it is appropriate to exercise enforcement discretion of violations Involving Special Circumstances in accordance with Section VII B.6 of the "General Statement of Policy and Procedures for NRC Enforcement Actions" (Enforcement Policy), NUREG 1600. Pending amendment to 10 CFR 70.24, further enforcement action will not be taken for failure to meet 10 CFR 70.24 provided an exemption to this regulation is obtained by NYPA before the next receipt of fresh fuel or before the next planned movement of fresh fuel at FitzPatrick. This item is tracked as non-cited violation (NCV 97-008-05).

V. MANAGEMENT MEETINGS

X1 Exit Meeting Summary

The inspectors presented the inspections results to members of the licensee management at the conclusion of the inspection on January 13, 1998. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

Licensee

G. Brownell, Licensing Engineer
M. Colomb, Site Executive Officer
D. Lindsey, General Manager, Operations
J. Maurer, General Manager, Support Services
A. McKeen, Radiological and Environmental Services Manager
T. Phelps, Radiological Supervisor
D. Ruddy, Director, Design Engineering
J. Solini, Sr. QA Engineer
D. Topley, General Manager, Maintenance
A. Zaremba, Licensing Manager

INSPECTION PROCEDURES USED

37551 Onsite Engineering

62707 Maintenance Observations

61726 Surveillance Observations

71707 Plant Operations

71750 Plant Support

83724 External Occupational Exposure Control and Personal Dosimetry

86750 Solid Radioactive Waste Management and Transportation of Radioactive Materials

92702 Follow-up on Corrective Actions for Violations and Deviations

TI 2515/133 Implementation of Revised 49 CFR Parts 100-179 and 10 CFR 71

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-333/97008-01	VIO	Improper performance of DC ground abnormal operating procedure resulted in HPCI logic actuation.
50-333/97008-02	VIO	Failure to enter Technical Specification Limiting Condition for Operation while troubleshooting electrical grounds
50-333/97008-03	VIO	Erroneously removal of HPCI components from the 10CFR50.49 environmental qualification program.
50-333/97008-04	IFI	Radioactive waste training program was not well organized and documented
50-333/97008-05	NCV	Failure to meet 10 CFR 70.24 requirements or to obtain a valid exemption from this regulation.

Closed

50-333/95006-03	URI	Components missing from control room ventilation drawings
50-333/96007-02	IFI	Affect of RWCU and CRD flow on alternate decay heat removal preoperational testing
50-333/97002-03	IFI	Adequacy of emergency procedures governing evacuation from areas near the new fuel storage vault and failure to meet requirements of 10 CFR 70.24 for new fuel criticality monitors
50-333/96007-08	VIO	Failure to follow plant Technical Specification for locked high radiation area entry
50-333/97008-05	NCV	Failure to meet 10 CFR 70.24 requirements or to obtain a valid exemption from this regulation.
EA 98-008	NCV	Adequacy of emergency procedures governing evacuation from areas near the new fuel storage vault and failure to meet requirements of 10 CFR 70.24 for new fuel criticality monitors.
50-333/96007-09	VIO	Failure to follow a formal quality assurance program

Discussed

None

LIST OF ACRONYMS USED

ADHR	Alternate Decay Heat Removal
AOP	Abnormal Operating Procedure
CFR	Code of Federal Regulations
CRD	Control Rod Drive
CST	Condensate Storage Tank
DC	Direct Current
DCM	Design Change Manual
DER	Deficiency & Event Report
DOT	Department of Transportation
EDG	Emergency Diesel Generator
EOF	Emergency Operations Facility
EQ	Environmental Qualification
ESF	Engineered Safety Feature
FE	Flow Element
FR	Federal Register
HIC	High Integrity Container
HPCI	High Pressure Coolant Injection
IFI	Inspection Follow-up Item
IR	Inspection Report
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LLRW	Low Level Radioactive Waste
LPCI	Low Pressure Coolant Injection
MOD	Motor Operated Damper
MOV	Motor Operated Valve
MP	Maintenance Procedure
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
OP	Operating Procedure
PCIS	Primary Containment Isolation System
PCP	Process Control Program
QA	Quality Assurance
QC	Quality Control
RAP	Reactor Analyst Procedure
RCA	Radiological Controlled Area
RP&C	Radiological Protection and Chemistry
RTID	Radiological Technical Information Document
RWCU	Reactor Water Clean-Up
SESO	System Engineer Standing Order
SNM	Special Nuclear Material
SRC	Safety Review Committee
SRV	Safety Relief Valve
SSC	Structures, Systems & Components
TI	Temporary Instruction
TS	Technical Specification

TVA	Tennessee Valley Authority
UFSAR	Updated Final Safety Analysis Report
VIO	Violation
WR	Work Request