February 13, 2008

Mr. Jack M. Davis
Senior Vice President and
Chief Nuclear Officer
Detroit Edison Company
Fermi 2 - 210 NOC
6400 North Dixie Highway
Newport, MI 48166

SUBJECT: FERMI POWER PLANT, UNIT 2, NRC INTEGRATED INSPECTION REPORT 05000341/2007006

Dear Mr. Davis:

On December 31, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Fermi Power Plant, Unit 2. The enclosed report documents the inspection findings which were discussed on January 17, 2008, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and to compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, eight findings of very low safety significance were identified, six of which involved violations of NRC requirements. However, because these findings were of very low safety significance and because the issues were entered into your corrective program, the NRC is treating these findings as Non-Cited Violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest the subject or severity of an Non-Cited Violation (NCV), you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Fermi 2 Facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Christine A. Lipa, Chief
Branch 4
Division of Reactor Projects

Docket No. 50-341
License No. NPF-43

Enclosure: Inspection Report 05000341/2007006
w/Attachment: Supplemental Information

cc w/encl: J. Plona, Vice President,
Nuclear Generation
K. Hlavaty, Plant Manager
R. Gaston, Manager, Nuclear Licensing
D. Pettinari, Legal Department
Michigan Department of Environmental Quality
Waste and Hazardous Materials Division
M. Yudasz, Jr., Director, Monroe County
Emergency Management Division
Supervisor - Electric Operators
State Liaison Officer, State of Michigan
Wayne County Emergency Management Division

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Letter to J. Davis from C. Lipa dated February 13, 2008

SUBJECT: FERMI POWER PLANT, UNIT 2, NRC INTEGRATED INSPECTION REPORT 05000341/2007006

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-341
License No:	NPF-43
Report No:	05000341/2007006
Licensee:	Detroit Edison Company
Facility:	Fermi Power Plant, Unit 2
Location:	Newport, Michigan
Dates:	October 1 through December 31, 2007
Inspectors:	R. Michael Morris, Senior Resident Inspector T. Steadham, Resident Inspector M. Bielby, Senior Operations Engineer K. Barclay, Reactor Engineer J. Cassidy, Health Physicist M. Franke, Senior Resident Inspector, Perry M. Holmberg, Reactor Inspector R. Jickling, Senior Emergency Preparedness Analyst R. Lerch, Project Engineer M. Munir, Reactor Engineer D. Reeser, Operations Engineer K. Streit, General Engineer A. Wilson, Reactor Engineer
Observers:	J. Moore, Environmental Project Manager / NSPDP A. Muñiz, NRR Project Manager
Approved by:	C. Lipa, Chief Branch 4 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000341/2007006; Detroit Edison Company, Fermi Power Plant, Unit 2; October 1 through December 31, 2007; In-Service Inspection Activities, Licensed Operator Requalification Program, Maintenance Risk Assessments and Emergent Work Control, Post-Maintenance Testing, Access Control to Radiologically Significant Areas, Follow up of Events and Notices of Enforcement Discretion.

This report covers a three-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Eight Green findings were identified, six of which were considered NCVs of NRC regulations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a Green NCV of 10 CFR 50.55a(b)5 for failure to perform additional pipe support examinations required by the American Society of Mechanical Engineers Code following adjustments to spring can settings on reactor water cleanup system support hanger G33-3096-G10. As a corrective action, the licensee performed a review of past pipe support examinations and identified ten similar examples (including support G33-3096-G10) and performed evaluations to confirm these pipe supports were operable.

  This finding was of more-than-minor significance because the finding could be reasonably viewed as a precursor to a significant event involving leakage from the reactor coolant system or attached support systems. In addition, the finding was associated with the Initiating Events Cornerstone attribute of "Equipment Performance," and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors applied the IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for the At-Power Situation," to this finding. The inspectors answered "no" to question 1 of the Initiating Events Cornerstone column of the phase 1 worksheet which asked, "Assuming worst case degradation, would the finding result in exceeding the Technical Specification limit for identified reactor coolant system leakage?" In this case, the worst case degradation would be leakage from fatigue cracks caused by inadequately supported piping. Because this issue was identified by the NRC prior to fatigue failure of Code piping components, this scenario did not occur. Therefore, the inspectors answered "no" to this question and the finding was determined to be of very low safety significance (Green). The primary cause of this finding was related to the cross-cutting area of Human Performance, Work Control Component (Item H.3(b) of IMC 0305) because the licensee failed to appropriately coordinate work activities between onsite work groups. (Section 1R08.1)

- Green. A self-revealed NCV of 10 CFR 50.65(a)(4) was identified for the failure to adequately assess the increased risk associated with the removal of shield blocks in the turbine building. After the licensee removed the shield blocks, the airflow in the turbine building was altered which caused steam tunnel temperatures to approach the main steam isolation valve closure trip set point. After operators identified the increased temperatures during routine rounds, the licensee installed a temporary barrier over the opening in the wall which restored normal ventilation flow throughout the building as immediate corrective actions. Consequently, steam tunnel temperatures returned to normal.

This finding was more than minor because had the risk assessment correctly recognized the potential for uncontrolled temperature increase leading to a plant trip, additional risk management actions that eventually needed to be taken would have been prescribed. This finding was of very low safety significance because although a reactor scram could have occurred, it had no affect on the availability of mitigation equipment or functions. The inspectors determined the finding was associated with cross-cutting aspect H.3(a), Human Performance, Work Control. (Section 1R13.1)

- Green. A self-revealed finding was identified for the failure to operate the plant in accordance with the documented instructions while performing a maintenance activity on the number 5 low pressure stop valve, which resulted in an unplanned power transient caused by the inadvertent opening of the main turbine bypass valves. The operators immediately stabilized the plant. Corrective actions taken by the licensee included performance management (coaching) of the involved personnel.

  This finding was more than minor because the failure to follow procedures led to an unplanned power transient. This finding was of very low safety significance because although an unplanned power transient occurred, it had no affect on the availability of mitigation equipment or functions. The inspectors determined the finding was associated with a cross-cutting aspect in the area of Human Performance, Work practices, because licensee personnel failed to follow written procedures when manipulating plant equipment (H.4(b)). No violation of NRC requirements occurred. (Section 1R19.1.b.(2))

- Green. A self-revealed NCV of Technical Specification 5.4.1.a was identified for the failure to perform a maintenance activity in accordance with the documented instructions which resulted in an unplanned scram from nine percent power. During a maintenance tagging activity, an operator inappropriately manipulated a valve in the open direction while verifying it was fully closed. The brief valve opening caused an unanticipated reactor water level reference leg pressure perturbation which was sensed as a drop in reactor water level, thereby initiating an alternate rod insertion on a reactor water level 2 signal. Operators immediately placed the mode switch in shutdown which completed the reactor scram. Corrective actions taken by the licensee included performance management (coaching) of the involved personnel and sharing the lessons learned from this event with other plant personnel.

  This finding was more than minor because the inappropriate valve operation led to a reactor scram. This finding was of very low safety significance because although a reactor scram occurred, the scram had no affect on the availability of mitigation equipment or functions. The inspectors determined the finding was associated with cross-cutting aspect H.4(b), Human Performance, Work Practices. (Section 4OA3.1)

- Green. A self-revealed NCV of 10 CFR 50.65(a)(4) was identified for the failure to appropriately assess the risk associated with removing the Reactor Protection System (RPS) from its normal power source. During refueling outage 12, the licensee placed RPS 'B' on its alternate power source to perform maintenance while RPS 'A' was out-of-service for maintenance. The licensee later started a residual heat removal pump while in this configuration which caused a voltage drop on the electrical bus powering RPS 'B'. The voltage drop caused RPS 'B' to trip on undervoltage which caused several system isolations including one system that was being relied upon as a method of decay heat removal. Corrective actions included revising the pertinent procedures to include a caution about not starting a residual heat removal pump while RPS was being powered from its alternate power source.

  This finding was more than minor because had the risk assessment correctly recognized the potential for the loss of a shutdown key safety function, additional risk management actions to preclude such a loss would have been prescribed. This finding was of very low safety significance because it did not affect the ability of operators to restore decay heat removal and the increase in average reactor coolant system temperature was negligible. (Section 4OA3.2)

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance was identified by the inspectors. The inspectors identified a performance deficiency based on licensed operators' failure to pass an NRC comprehensive biennial written examination. Of the 48 licensed operators evaluated, 10 did not pass their required biennial written examination.

  The finding was more than minor because it reflects potential shortcomings in the ability to conduct routine operation/maintenance and respond to actual abnormal or emergency conditions. The finding is of very low safety significance because the operators were removed from watch-standing duties during the period in which the annual testing of the operators was conducted, there were no actual consequences due to the failures, and the associated operators were retrained and re-evaluated before they were authorized to resume the performance of licensed operator duties. Based on the licensee's successful remediation and subsequent re-testing of individuals who failed the examinations, no violation of regulatory requirements occurred. (Section 1R11.10)

- Green. A self-revealed NCV of 10 CFR 50, Appendix B, Criterion III, "Design Control," was identified for the failure to properly maintain design control of the reactor core isolation cooling pump. The licensee replaced all four pump impellers in 1996 but failed to ensure the design documentation correctly identified the actual installed impeller diameters. As a result, one of the installed impellers was larger than specified on the design documents. When the licensee procured a new rotating assembly for installation in refueling outage 12, those design documents were used to specify the replacement impeller sizes. Consequently, one of the four replacement impellers was smaller than the one it replaced and the pump would not have been able to develop sufficient head to perform its intended safety function. Once identified, the licensee reinstalled the old rotating assembly and updated the design documents accordingly as immediate corrective actions.

  This finding was more than minor because the incorrect design documents were used to procure an inadequately-sized impeller; the impeller was later installed in the pump; the undersized impeller adversely affected the operation of the pump; and the pump was returned to service prior to discovery of the problem. This finding was of very low safety significance because it did not represent an actual loss of safety function for greater than its Technical Specification allowable outage time because high pressure coolant injection remained available during all relevant periods. (Section 1R19.1.b.(1))

Cornerstone: Occupational Radiation Safety

- Green. A self-revealed finding of very low safety significance and an associated NCV of regulatory requirements was identified for workers entering a high radiation area without an adequate awareness of radiological conditions and without performing work using a radiation work permit that allowed entry into a high radiation area. The electronic dosimetry worn by one of the workers alarmed when elevated dose rates were encountered. Corrective actions taken by the licensee included performance management (coaching) of the involved personnel. The licensee also performed additional communications to the plant population through informational notices and a site stand-down to reinforce that all workers ensure they read all posted signs and work on the correct radiation work permits.

  The issue was more than minor because it was associated with the Program/Process attribute of the Occupational Radiation Safety Cornerstone and affected the cornerstone objective to ensure adequate protection of worker health and safety from exposure to radiation. The issue represents a finding of very low safety significance because it did not involve as-low-as-is-reasonably-achievable (ALARA) planning or work controls, there was no overexposure or substantial potential for an overexposure given the radiological conditions in the area, nor was the licensee's ability to assess worker dose compromised. An NCV of Technical Specification 5.7.1.b was identified for entering a high radiation area without an adequate awareness of radiological conditions and without a radiation work permit that allowed access into a high radiation area. Additionally, this finding has a cross-cutting aspect in the area of Human Performance because the radiation protection technician did not validate the intended work area and the workers did not perform a self check or peer check of the requirements needed before entering the high radiation area. (H.4(a)). (Section 2OS1.3)

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

Summary of Plant Status

Unit 2 was shutdown in refueling outage 12 (RF12) at the beginning of this inspection period. The plant began start-up following the refueling outage on November 14. A reactor scram occurred on November 15 from nine percent power. The plant resumed power operations on November 18. Down powers to 60 percent were performed on November 20, November 24, and December 1, for rod pattern adjustments. The plant operated at or near full power the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Winter Seasonal Readiness Preparations

a. Inspection Scope

The inspectors conducted a review of the licensee's preparations for winter conditions to verify the plant's design features and implementation of procedures were sufficient to protect mitigating systems from the effects of adverse weather. Documentation for selected risk-significant systems was reviewed to ensure these systems would remain functional when challenged by inclement weather. During the inspection, the inspectors focused on plant-specific design features and the licensee's procedures used to mitigate or respond to adverse weather conditions. Additionally, the inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) and performance requirements for systems selected for inspection and verified operator actions were appropriate as specified by plant specific procedures. Cold weather protection, such as area heaters, was verified to be in operation where applicable. Specific documents reviewed during this inspection are listed in the Attachment. The inspectors' reviews focused specifically on the following plant systems due to their risk significance or susceptibility to cold weather issues:

- Residual Heat Removal (RHR) Complex;
- Cooling Tower Isolation Valves; and
- General Service Water Building.

These inspection activities constitute one winter seasonal readiness preparations sample as defined in Inspection Procedure 71111.01-05.

b. Findings

No findings of significance were identified.

.2 Readiness for Impending Adverse Weather Condition – Tornado Watch

a. Inspection Scope

Since thunderstorms with potential tornados and high winds were forecast in the vicinity of the facility for October 18, 2007, the inspectors reviewed the licensee's overall preparations/protection for the expected weather conditions. On October 18, 2007, the inspectors walked down the Division II switchyard and vicinity, the reactor building refueling floor, and the licensee's emergency alternating current (AC) power systems because their safety-related functions could be affected or required as a result of high winds or tornado-generated missiles or the loss-of-offsite power. The inspectors evaluated the licensee's preparations against the site's procedures and determined the staff's actions were adequate. During the inspection, the inspectors focused on plant specific design features and the licensee's procedures used to respond to specified adverse weather conditions. The inspectors also toured the plant grounds for loose debris which could become missiles during a tornado and ascertained operator staffing and if they could access controls and indications for those systems required to control the plant. Additionally, the inspectors reviewed the UFSAR and performance requirements for systems selected for inspection and verified operator actions were appropriate as specified by plant specific procedures. The inspectors also reviewed corrective action program items to verify the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their corrective action program in accordance with station corrective action procedures. Specific documents reviewed during this inspection are listed in the Attachment.

These inspection activities constitute one readiness for impending adverse weather condition sample as defined in Inspection Procedure 71111.01-05.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns (71111.04Q)

a. Inspection Scope

The inspectors performed a partial system walkdown of the following risk-significant system:

- Emergency Diesel Generator (EDG) Fuel Oil System.

The inspectors selected this system based on its risk significance relative to the Reactor Safety Cornerstone at the time they were inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, UFSAR, Technical Specification (TS) requirements, Administrative TS, outstanding work orders, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify there were no obvious deficiencies. The inspectors also verified the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program with the appropriate significance characterization. Documents reviewed are listed in the Attachment.

These inspection activities constitute one partial system walkdown sample as defined in Inspection Procedure 71111.04-05.

b. Findings

No findings of significance were identified.

.2 Semi-Annual Complete System Walkdown (71111.04S)

a. Inspection Scope

On October 16, 2007, the inspectors performed a complete system alignment inspection of the RHR system to verify the functional capability of the system. This system was selected because it was considered both safety significant and risk significant in the licensee's probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment line ups, electrical power availability, system pressure and temperature indications as appropriate, component labeling, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of past and outstanding work orders (WO) was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the corrective action resolution document (CARD) database to ensure system equipment alignment problems were being identified and appropriately resolved. The documents used for the walkdown and issue review are listed in the Attachment.

These inspection activities constitute one complete system walkdown sample as defined in Inspection Procedure 71111.04-05.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

The inspectors conducted fire protection walkdowns which were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- Drywell;
- Main Transformer 2A Deluge System Walkdown and Test;
- Reactor Building, First Floor; and
- Reactor Building, Fourth Floor.

The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and had implemented adequate compensatory measures for out-of-service, degraded or inoperable fire protection equipment, systems, or features in accordance with the licensee's fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events with later additional insights, their potential to impact equipment which could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Using the documents listed in the Attachment, the inspectors verified fire hoses and extinguishers were in their designated locations and available for immediate use; fire detectors and sprinklers were unobstructed; transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified minor issues identified during the inspection were entered into the licensee's corrective action program.

These inspection activities constitute four quarterly fire protection inspection samples as defined in Inspection Procedure 71111.05-05.

b. <u>Findings</u>

No findings of significance were identified.

1R06 <u>Flooding</u> (71111.06)

.1 <u>Internal Flooding</u>

a. <u>Inspection Scope</u>

The inspectors reviewed selected risk important plant design features and licensee procedures intended to protect the plant and its safety-related equipment from internal flooding events. The inspectors reviewed flood analyses and design documents, including the UFSAR, engineering calculations, and abnormal operating procedures, for licensee commitments. The specific documents reviewed are listed in the Attachment. In addition, the inspectors reviewed licensee drawings to identify areas and equipment that may be affected by internal flooding caused by the failure or misalignment of nearby sources of water, such as the fire suppression or the circulating water systems. The inspectors also reviewed the licensee's corrective action documents with respect to past flood-related items identified in the corrective action program to verify the adequacy of the corrective actions. The inspectors performed a walkdown of the following plant areas to assess the adequacy of watertight doors and to verify drains and sumps were clear of debris and operable and that the licensee complied with its commitments:

- RHR Complex and Auxiliary Building First Floor Cable Tunnel including the Review of Operating Experience Smart Sample FY2007-02, related to IN 2005-30 and issues associated with conduit/hydrostatic seal issues."

These inspection activities constitute one internal flooding sample as defined in Inspection Procedure 71111.06-05.

a. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

.1 Heat Sink Performance

a. Inspection Scope

The inspectors reviewed the licensee's testing of EDG-12 lubricating oil, jacket coolant, and air coolant heat exchangers to verify potential deficiencies did not mask the licensee's ability to detect degraded performance, to identify any common-cause issues that had the potential to increase risk, and to ensure the licensee was adequately addressing problems that could result in initiating events that would cause an increase in risk. The inspectors reviewed the licensee's observations as compared against acceptance criteria, the correlation of scheduled testing, the frequency of testing, and the impact of instrument inaccuracies on test results. Inspectors also verified that test acceptance criteria considered differences between test conditions, design conditions, and testing criteria.

These inspection activities constitute one annual heat sink performance sample as defined in Inspection Procedure 71111.07-05.

b. Findings

No findings of significance were identified.

1R08 In-Service Inspection Activities (71111.08)

.1 Piping Systems In-Service Inspection

a. Inspection Scope

From October 9 through 12, 2007, the inspectors conducted a review of the implementation of the licensee's in-service inspection (ISI) program for monitoring degradation of the reactor coolant system boundary and the risk-significant piping system boundaries. The inspectors selected the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI required examinations and Code components in order of risk priority as identified in Section 71111.08-03 of the inspection procedure, based upon the ISI activities available for review during the onsite inspection period.

The inspectors observed manual ultrasonic examination of the recirculation loop ‘A’ suction nozzle (N1A) safe-end-to-pipe weld FW-RS-2-A1 to evaluate compliance with the ASME Code Section XI requirements and to verify that indications and defects (if present) were dispositioned in accordance with the ASME Code Section XI.

The inspectors observed dye penetrant examination of control rod drive (CRD) housing weld CRDH-X02-Y39-W2 to evaluate compliance with the ASME Code Section XI and Section V requirements and to verify that indications and defects (if present) were dispositioned in accordance with the ASME Code Section XI requirements.

The inspectors reviewed recordable indications identified during ultrasonic examination of the reactor pressure vessel shell-to-flange weld 13-308 and during visual VT-3 examinations of piping supports to determine if the licensee’s corrective actions and extent-of-condition reviews were in accordance with the ASME Code Section XI requirements.

The inspectors reviewed pressure boundary weld records for replacement of RHR pump ‘B’ discharge check valve E1100 F031B to determine if the welding acceptance and pre-service examination records (e.g., radiography, pressure testing, visual, dye penetrant, and weld procedure qualification tensile tests and bend tests) were in accordance with ASME Code Sections III, V, IX, and XI requirements.

The inspectors performed a review of ISI-related problems that were identified by the licensee and entered into the corrective action program, conducted interviews with licensee staff, and reviewed licensee corrective action records to determine if:

- the licensee had described the scope of the ISI-related problems;
- the licensee had established an appropriate threshold for identifying issues;
- the licensee had evaluated industry generic issues related to ISI and pressure boundary integrity; and
- the licensee implemented appropriate corrective actions.

The inspectors performed these reviews to ensure compliance with 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” requirements. The corrective action documents reviewed by the inspectors are listed in the Attachment to this report.

These inspection activities constitute one in-service inspection sample as defined in Inspection Procedure 71111.08.

b. Findings

Failure to Expand Scope of Pipe Support Hanger Examinations

Introduction: The inspectors identified a Green NCV of 10 CFR 50.55a(b)5 for failure to perform additional pipe support examinations required by the ASME Code following adjustments to spring can settings on a reactor water cleanup (RWCU) system support.

Description: On September 19, 2007, the inspectors identified that the licensee failed to perform examinations of adjacent RWCU supports following adjustments made to the spring can settings of support hanger G33-3096-G10.

Support hanger G33-3096-G10 is located on the 6-inch diameter RWCU piping near the reactor vessel lower head drain isolation valve G3352 F001. In 1996, the licensee issued Engineering Design Package (EDP) 26858 to replace the motor operator of valve G3352 F001 with a larger motor operator. To support the additional weight on the RWCU pipe, the two spring can settings for support hanger G33-3096-G10 were increased to 935 pounds and the two spring can settings for support hanger G33-3096-12 were increased to 640 pounds (cold settings). However, the licensee failed to revise the support drawings for these hangers in 1996 to incorporate the new spring can settings required by this EDP.

On April 4, 2006, during an ASME Code VT-3 examination of RWCU system support hanger G33-3096-G10, the licensee examiner identified that the hanger spring cans were set at 950 pounds instead of 810 pounds for the cold setting as identified on drawing PIS G33-52-G. The licensee reset these spring cans to 810 pounds (incorrect setting) and returned the system to service on April 11, 2006. On September 19, 2007, the NRC inspectors questioned the reason for this support adjustment, which prompted the licensee staff to identify the incorrect spring can setting on drawing PIS G33-52-G. The licensee entered this issue into the corrective action program (CARD 07-25257) and initiated actions to correct the spring can support settings. The inspectors determined this issue was a violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," of minor significance because the licensee concluded that past plant operation with the incorrect spring can setting did not affect operability of support hanger G33-3096-G10.

On April 4, 2006, following VT-3 examination of RWCU system support hanger G33-3096-G10, the licensee examiner identified the discrepant spring can setting, documented this condition in CARD 06-22493, and notified the Component Support Engineer. However, the Component Support Engineer did not initiate an ISI-NDE evaluation to evaluate Code examination expansion requirements or contact Plant Support Engineering to assist in performing an operability evaluation as required by step 5.3.4 of Procedure 43.000.004, "Visual Examination of Component Supports." Consequently, the licensee failed to perform examinations of component supports adjacent to support G33-3096-G10 prior to returning the RWCU system to service on April 11, 2006. This error occurred because the Component Support Engineer misinterpreted step 2.1 of ISI-NDE Program, "In-service Inspection - Nondestructive Examination Program (Plan) for Component Supports," which stated, "Component supports determined not to have integrity for intended service are defined as inoperable. Additional examinations as defined per Code Case N-491-1, paragraph 2430, for component supports determined to be inoperable." Because the Component Support Lead Engineer did not consider the out-of-adjustment spring cans to be an inoperable condition, an ISI-NDE evaluation was not initiated to evaluate the need for expanding the scope of support examinations on the RWCU system. In this case, the ASME Code requirement for expansion of VT-3 examinations is based upon the corrective measures required (e.g., spring can adjustments) and not based on a determination of support operability. The licensee entered this issue into the corrective action program (CARD 07-25293) and initiated actions to review past VT-3 examinations of component supports. During this review, the licensee identified ten similar examples (including support G33-3096-G10) and performed evaluations to confirm these pipe supports were operable.

<u>Analysis</u>: The inspectors determined the failure of the licensee to perform the additional pipe support examinations required by the Code was a performance deficiency that

warranted a significance evaluation. Absent NRC intervention, the licensee staff would not have performed additional pipe support examinations due to a misinterpretation of when to apply the Code examination expansion requirements. Failure to expand Code pipe support examinations could place the reactor coolant pressure boundary and attached support systems at an increased risk for leakage or failure from fatigue cracks caused by inadequately supported piping. Therefore, this finding was of more-than-minor significance because the finding could be reasonably viewed as a precursor to a significant event involving leakage from the reactor coolant system or attached support systems. In addition, the finding was associated with the Initiating Events Cornerstone attribute of “Equipment Performance” and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations.

The inspectors applied IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for the At-Power Situation,” to this finding. The inspectors answered “no” to question 1 of the Initiating Events Cornerstone column of the Phase 1 worksheet, which asked, “Assuming worst case degradation, would the finding result in exceeding the TS limit for identified reactor coolant system (RCS) leakage?” In this case, the worst case degradation would be leakage from fatigue cracks caused by inadequately supported piping. Because this issue was identified by the NRC prior to fatigue failure of Code piping components, this scenario did not occur. Therefore, the inspectors answered “no” to this question and the finding was determined to be of very low safety significance (Green).

The primary cause of this finding was related to the cross-cutting area of Human Performance, Work Control Component, (Item H.3(b) of IMC 305) because the licensee failed to appropriately coordinate work activities between onsite work groups. Specifically, the Component Support Lead Engineer failed to coordinate results of the VT-3 examinations with the ISI lead or Plant Support Engineering.

<u>Enforcement</u>: On October 12, 2007, the inspectors identified an NCV of 10 CFR 50.55a(b)5. Title 10 CFR 50.55a(b)5 required in part that licensees may apply the ASME Boiler and Pressure Vessel Code cases listed in Regulatory Guide 1.147 without prior NRC approval. Regulatory Guide 1.147 identified Code Case N-491-1 as approved.

The licensee applied Code Case N-491-1 as identified in step 2.1 of ISI-NDE Program, “In-service Inspection - Nondestructive Examination Program (Plan) for Component Supports” Revision 5. Paragraph 3122.2 of Code Case N-491-1 described corrective measures which included adjustment of improper hot or cold settings of spring supports and constant load supports. Paragraph 2430(a) of Code Case N-491-1 required in part that when component supports must be subject to corrective measures in accordance with -3000, the component supports immediately adjacent to those for which corrective action is required shall be examined.

Contrary to the above, on April 19, 2006, corrective measures (adjustment to the cold spring can settings) were made to support hanger G33-3096-G10 without examination of the immediately adjacent component supports. Failure to perform examination of the adjacent supports is a violation of 10 CFR 50.55a(b)5. Because of the very low safety significance of this finding and because the issue was entered into the licensee’s corrective action program (CARD 07-25293), this finding is being treated as an NCV, consistent with Section VI.A.1 of the Enforcement Policy: NCV 05000341/2007006-01, Failure to Expand Scope of Pipe Support Hanger Examinations.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On November 27, 2007, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification examinations to verify operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of abnormal and emergency procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements.

These inspection activities constitute one quarterly licensed operator requalification program sample as defined in Inspection Procedure 71111.11.

b. Findings

No findings of significance were identified.

Biennial Licensed Operator Requalification Program Inspection (71111.11B)

.2 Facility Operating History

a. Inspection Scope

The inspectors reviewed the plant's operating history from December 2005 through November 2007 to identify operating experience that was expected to be addressed by the Licensed Operator Requalification Training (LORT) program. The inspector verified that the identified operating experience had been addressed by the facility licensee in accordance with the station's approved Systems Approach to Training (SAT) program to satisfy the requirements of 10 CFR 55.59(c). The documents reviewed during this inspection activity are listed in the Attachment.

b. Findings

No findings of significance were identified.

.3 Licensee Requalification Examinations

a. Inspection Scope

The inspectors performed an inspection of the licensee's LORT test/examination program for compliance with the station's SAT program which would satisfy the requirements of 10 CFR 55.59(c)(4). The reviewed operating examination material consisted of six operating tests, each containing two dynamic simulator scenarios and five job performance measures. The written examinations reviewed consisted of six written examinations, each including 32 questions. The inspectors reviewed a sample of the annual requalification operating test and biennial written examination material to evaluate general quality, construction, and difficulty level. The inspectors assessed the level of operating examination material duplication from week to week during 2007. The examiners assessed the amount of written examination material duplication from week to week for the written examination administered in 2007. The inspectors reviewed the methodology for developing the examinations including the LORT program 2-year sample plan, probabilistic risk assessment insights, previously identified operator performance deficiencies, and plant modifications. The documents reviewed during this inspection activity are listed in the Attachment.

b. Findings

No findings of significance were identified.

.4 Licensee Administration of Requalification Examinations

a. Inspection Scope

The inspectors observed the administration of a requalification operating test to assess the licensee's effectiveness in conducting the test to ensure compliance with 10 CFR 55.59(c)(4). The inspectors evaluated the performance of one crew in parallel with the facility evaluators during two dynamic simulator scenarios and evaluated various licensed crew members concurrently with facility evaluators during the administration of several job performance measures. The inspectors assessed the facility evaluators' ability to determine adequate crew and individual performance using objective, measurable standards. The inspectors observed the training staff personnel administer the operating test including conducting pre-examination briefings, evaluations of operator performance, and individual and crew evaluations upon completion of the operating test. The inspectors evaluated the ability of the simulator to support the examinations. A specific evaluation of simulator performance was conducted and documented under Section 1R11.8, "Conformance with Simulator Requirements Specified in 10 CFR 55.46," of this report. The documents reviewed during this inspection activity are listed in the Attachment.

The inspectors reviewed a sample of the graded biennial written examinations to assess the licensee's ability to use performance standards to consistently and objectively evaluate individual performance.

b. Findings

Introduction: During the conduct of one of the dynamic simulator scenarios the inspectors identified an unresolved item related to a procedure change.

Description: During one of the dynamic simulator scenarios, conditions were simulated during an Anticipated Transient Without Scram (ATWS) that required the operating crew to lower reactor pressure vessel (RPV) water level in accordance with emergency operating procedure (EOP) 29.100.01, Sheet 1A, "RPV Control-ATWS." The purpose of lowering RPV water level is to reduce core inlet sub-cooling and thus reduce the potential for power oscillations. EOP 29.100.01, Sheet 1A, directs the operators to "Terminate and Prevent" all injection flow into the RPV except for flow from the CRD, Reactor Core Isolation Cooling (RCIC), and Standby Liquid Control (Boron) systems. Contrary to the BWR Owners' Group (BWROG) Emergency Procedure Guidelines (EPG) and Severe Accident Guidelines [Revision 2] – which states that failure to completely stop RPV injection flow (with the exception of CRD, RCIC, and Standby Liquid Control) would delay the reduction in core inlet sub-cooling, thus increasing the potential for flux oscillations – the crew was observed to implement this step (FSL-10), in accordance with the licensee's expectations, by turning OFF the low pressure Emergency Core Cooling Systems (ECCS) and Standby Feedwater pumps, reducing High Pressure Coolant Injection flow to 0 gpm, and reducing (i.e., NOT stopping) Feedwater system flow so that level decreased in a controlled manner. When asked why the licensee's procedural steps deviated from the BWROG EPG, they stated that the deviation was necessary to allow time for bypassing of interlocks to prevent the loss of the Main Condenser heat sink, and to prevent dropping water level below the top of active fuel. The BWROG EPG states that reducing reactor power and preventing power oscillations is of greater importance than preventing loss of the main condenser.

Technical Specification 5.4.1 requires, in part, that written procedures/instructions be established, implemented, and maintained covering the emergency operating procedures required to implement the requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements," and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33. NUREG-0737 and the associated Supplement 1 required licensees to analyze transients and accidents, prepare emergency procedure technical guidelines, and develop symptom-based emergency operating procedures based on those technical guidelines. The BWROG EPG provides the technical basis for the development of the emergency operating procedures used by BWR licensees. Licensees are permitted to deviate from the BWROG guidelines provided they document the technical basis for the deviation. When asked to provide justification for the deviation from the BWROG EPG, the licensee was unable to do so. The licensee has initiated action (CARD 07-28195), through their corrective action program, to provide the necessary basis for the deviation.

This issue is an Unresolved Item (URI) pending further NRC review and completion of the licensee's actions to provide the necessary documentation to support the deviation: URI 05000341/2007006-02, Undocumented Technical Basis for Change to EOP ATWS Mitigation Strategy.

.5 Examination Security

a. Inspection Scope

The inspectors observed and reviewed the licensee's overall licensed operator requalification examination security program related to examination physical security (e.g., access restrictions and simulator considerations) and integrity (e.g., predictability and bias) to verify compliance with 10 CFR 55.49, "Integrity of Examinations and Tests." The inspectors also reviewed the facility licensee's examination security procedure, any corrective actions related to past or present examination security problems at the facility, and the implementation of security and integrity measures (e.g., security agreements, sampling criteria, bank use, and test item repetition) throughout the examination process. The documents reviewed during this inspection activity are listed in the Attachment.

b. Findings

No findings of significance were identified.

.6 Licensee Training Feedback System

a. Inspection Scope

The inspectors assessed the methods and effectiveness of the licensee's processes for revising and maintaining its LORT Program up to date including the use of feedback from plant events and industry experience information. The inspectors reviewed the licensee's quality assurance oversight activities, including licensee training department self-assessment reports. The inspectors evaluated the licensee's ability to assess the effectiveness of its LORT program and their ability to implement appropriate corrective actions. This evaluation was performed to verify compliance with 10 CFR 55.59(c) and the licensee's SAT program. The documents reviewed during this inspection activity are listed in the Attachment.

b. Findings

No findings of significance were identified.

.7 Licensee Remedial Training Program

a. Inspection Scope

The inspectors assessed the adequacy and effectiveness of the remedial training conducted since the previous biennial requalification examinations and the training from the current examination cycle to ensure they addressed weaknesses in licensed operator or crew performance identified during training and plant operations. The inspectors reviewed remedial training procedures and individual remedial training plans. This evaluation was performed in accordance with 10 CFR 55.59(c) and with respect to the licensee's SAT program. The documents reviewed during this inspection activity are listed in the Attachment.

b. Findings

No findings of significance were identified.

.8 Conformance with Operator License Conditions

a. Inspection Scope

The inspectors reviewed the facility and individual operator licensees' conformance with the requirements of 10 CFR 55. The inspectors reviewed the facility licensee's program for maintaining active operator licenses to assess compliance with 10 CFR 55.53(e) and (f). The inspectors reviewed the procedural guidance and the process for tracking on-shift hours for licensed operators and which control room positions were granted watch-standing credit for maintaining active operator licenses. The inspectors reviewed the facility licensee's LORT program to assess compliance with the requalification program requirements as described by 10 CFR 55.59(c). Additionally, medical records for six licensed operators were reviewed for compliance with 10 CFR 55.53(I). The documents reviewed during this inspection activity are listed in the Attachment.

b. Findings

No findings of significance were identified.

.9 Conformance with Simulator Requirements Specified in 10 CFR 55.46

a. Inspection Scope

The inspectors assessed the adequacy of the licensee's simulation facility (simulator) for use in operator licensing examinations and for satisfying experience requirements as prescribed in 10 CFR 55.46, "Simulation Facilities." The inspectors also reviewed a sample of simulator performance test records (i.e., transient tests, malfunction tests, steady state tests, and core performance tests), simulator discrepancies, and the process for ensuring continued assurance of simulator fidelity in accordance with 10 CFR 55.46. The inspectors reviewed and evaluated the discrepancy process to ensure simulator fidelity was maintained. Open simulator discrepancies were reviewed for importance relative to the impact on 10 CFR 55.45 and 55.59 operator actions as well as on nuclear and thermal hydraulic operating characteristics. The inspectors conducted interviews with members of the licensee's simulator staff about the configuration control process and completed the IP 71111.11, Appendix C, checklist to evaluate whether or not the licensee's plant-referenced simulator was operating adequately as required by 10 CFR 55.46(c) and (d). The documents reviewed during this inspection activity are listed in the Attachment.

b. Findings

No findings of significance were identified.

.10 Biennial Written Examination and Annual Operating Test Results

a. Inspection Scope

The inspectors reviewed the overall performance of licensed operators to maintain their license by successfully completing a requalification program. Pass/fail results of the biennial comprehensive written examination and the annual job performance measure and simulator operating tests (required to be given per 10 CFR 55.59(a)(2)) administered by the licensee in November and December 2007 were reviewed. These results were compared to the thresholds established in IMC 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process." The documents reviewed during this inspection activity are listed in the Attachment.

b. Findings

(1) Individual Operator Performance on the Biennial Written Examination Portion of the 2007 Facility-Administered Annual Requalification Examination

Introduction: The inspectors identified a finding of very low safety significance (Green) associated with more than 20 percent of the licensed operators not passing the facility-administered biennial written examination portion of the annual requalification examination. Based on the licensee's successful remediation and subsequent re-testing of individuals who failed the examinations, no violation of regulatory requirements occurred.

Description: During November and December 2007, the facility licensee's training staff administered the biennial comprehensive written examination, required by 10 CFR 55.59(a)(2), to assess individual licensed operator knowledge and abilities needed to operate the facility. The written examination was developed using the facility licensee's performance standards, derived from NUREG-1021, "Operator Licensing Examination Standards for Power Reactors." The results from the grading of the written examinations revealed that 10 of the 48 individuals evaluated failed to pass the administered written examinations. In accordance with their SAT-based licensed operator requalification training program, the licensee removed the individuals from shift and conducted successful remediation and re-testing of the individuals who failed the examinations. The facility licensee initiated CARD 08-20103 to investigate the cause for the high number of examination failures, after the inspectors informed the licensee that the failure rate exceeded the threshold for an inspection finding.

Analysis: A performance deficiency was identified based on licensed operators' failure to pass an NRC comprehensive biennial written examination in that 10 out of the 48 licensed operators evaluated failed to pass a comprehensive requalification written examination as required by 10 CFR 55.59(a)(2). These 10 operators failed to demonstrate through their performance on the examination a satisfactory understanding of the knowledge and abilities needed to safely operate the facility. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for affecting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements or licensee procedures. The finding is greater than minor because the performance deficiency potentially affects the Human Performance attributes of the Initiating Events, Mitigating Systems, and Barrier Integrity Cornerstone objectives to:

- limit the likelihood of events that upset plant stability and challenge critical safety functions;
- ensure mitigating system availability, reliability, and capability to respond to initiating events to prevent undesirable consequences; and
- provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by events or accidents.

Specifically, the finding reflects potential shortcomings in the ability to conduct routine operation/maintenance and respond to actual abnormal or emergency conditions. The risk, associated with the number of licensed operators not passing the biennial written examination is provided by the SDP Flow Chart found in IMC 0609, Appendix I, "Operator Requalification Human Performance SDP." Based on a 20.8 percent written examination failure rate (10 failures out of the 48 licensed operators who were administered the biennial written examination), the finding was characterized by the SDP as having a very low safety significance, or Green.

Enforcement: NRC regulations require the facility licensee to develop, implement, and maintain a requalification program that ensures the facility's licensed operators maintain the knowledge and abilities needed to safely operate the facility. Mastery of the training program objectives is demonstrated by successful completion of a comprehensive biennial written examination as well as an annual operating test as required by NRC regulations. When an examination failure occurs, requirements are met by restricting the associated operator from performing licensed duties until the operator has been retrained and successfully re-tested - steps which the facility licensee has completed. Therefore, no violation of regulatory requirements occurred. Operator performance on the 2007 biennial written examination has been entered into the licensee's corrective action program as CARD 08-20103 and the licensee is performing an assessment to determine the cause for the high number individual licensed operator failures. This finding is described as FIN 05000341/2007006-03 (Individual Operator Performance on the Biennial Written Examination Portion of the 2007 Facility-Administered Annual Requalification Examination)

(2) Individual Operator/Crew Performance on the Annual Operating Test

No findings of significance were identified for individual operator or crew performance on the annual operating test portion of the facility-administered annual requalification examination.

.11 Completion Status

The inspection activities described in Sections 1R11.2 through 1R11.10 constitute one biennial licensed operator requalification program inspection sample.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations (71111.12Q)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving governor oscillations on the Emergency Diesel Generators.

The inspectors reviewed events where ineffective equipment maintenance could have resulted in degraded performance and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- implementing appropriate work practices;
- identifying and addressing common-cause failures;
- scoping of systems in accordance with 10 CFR 50.65(b) of the maintenance rule;
- characterizing system reliability issues for performance;
- charging unavailability for performance;
- trending key parameters for condition monitoring;
- ensuring 10 CFR 50.65(a)(1) or (a)(2) classification or re-classification; and
- verifying appropriate performance criteria for structures, systems, and components/functions classified as (a)(2) or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the corrective action program with the appropriate significance characterization. Documents reviewed are listed in the Attachment.

These inspection activities constitute one quarterly maintenance effectiveness sample as defined in Inspection Procedure 71111.12-05.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

.1 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify the appropriate risk assessments were performed:

- division I load sequencer failure during the week of October 1;
- shutdown cooling outage, division swap, and division I outage during the week of October 15;
- unplanned RCIC impeller replacement during the week of November 26; and
- turbine building shield block removal during the week of November 26.

These activities were selected based on their potential risk significance relative to the Reactor Safety Cornerstones. As applicable for each activity, the inspectors verified risk assessments were performed as required by 10 CFR 50.65(a)(4) and were accurate and complete. When emergent work was performed, the inspectors verified the plant risk was promptly reassessed and managed. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met.

These inspection activities constitute four maintenance risk assessments and emergent work control samples as defined in Inspection Procedure 71111.13-05.

b. Findings

Inadequate Work Planning for Shield Block Wall Removal

Introduction: A Green self-revealed NCV of 10 CFR 50.65(a)(4) was identified for the failure to manage the increase in risk prior to removing a shield block wall for the turbine building steam tunnel. As a result, cooling to the steam tunnel was adversely affected which caused the temperature to increase and approach the set point for isolating the main steam isolation valves (MSIVs).

Description: On September 19, 2007, during routine rounds, operators noticed elevated turbine building steam tunnel temperatures. The relevant temperature indicators were reading approximately 190 degrees Fahrenheit (^{O}F), which was approximately 20^{O} F higher than expected. Operators became concerned at the elevated temperatures because they could have been indicative of a steam leak and because the set point for automatic MSIVs isolation was 200^{O} F. Consequently, operators began hourly temperature monitoring and began investigating the cause of the elevated temperatures.

Based on observations of the change in airflow in the turbine building, observations of other system parameters, and a review of plant drawings, the operators concluded that steam tunnel cooling had been adversely impacted as a result of the change in ventilation flow path resulting from the shield block wall removal. The normal major turbine building ventilation flow path was from the upper elevations of the turbine building, past the condenser, through the steam tunnel, to the basement, and out through the exhaust stack. With the shield block wall removed, the air flow through the second floor steam tunnel was reduced sufficiently to cause temperatures in that area to increase and approach the MSIV trip setpoint.

Licensee Procedure 23.412, Revision 45, “Turbine Building Heating, Ventilation, and Air Conditioning System,” (TBHVAC) precaution and limitation 3.5 states that if the turbine building fans are off for more than a few minutes with the reactor at power, the temperature of the stagnant air in the turbine building steam tunnel could exceed 200^{O} F and cause a Group 1 isolation and reactor scram. Because the temperature probes that provide an input to the MSIV isolation logic are located in this area and the temperature in the area reached 190^{O} F and continued to rise, the inspectors concluded that the shield block wall removal increased the likelihood of a plant transient.

The licensee implemented compensatory measures which included hourly monitoring of the second floor steam tunnel temperatures and the installation of a temporary barrier over the first floor steam tunnel opening to restore the normal ventilation flow through the turbine building. After the barrier was installed, operators immediately noticed a reduction in the steam tunnel temperature indicators.

Analysis: The inspectors determined the failure to perform an adequate risk assessment of the shield block wall removal was a performance deficiency. The inspectors determined the finding was more than minor in accordance with IMC 0612, Appendix B, Section 3, dated September 30, 2005. The inspectors compared this issue with the examples contained in Appendix E of IMC 0612 and determined that example 7.f is related to this issue. From this example, the inspectors concluded the issue was more than minor because had the risk assessment correctly recognized the potential for uncontrolled temperature increase leading to a plant trip, additional risk management actions that eventually needed to be taken would have been prescribed.

The inspectors determined this finding affected the Initiating Events Cornerstone as a transient initiator. The inspectors performed a Phase 1 screening of this issue and determined that because the finding increased the likelihood of a full MSIV closure, the likelihood of both a reactor trip and that mitigating equipment or functions would not be available increased. Therefore, the inspectors performed a Phase 2 analysis.

The inspectors used the Risk-Informed Inspection Notebook for Fermi 2 Nuclear Power Plant, Revision 2, dated September 30, 2005. The inspectors determined that this finding was related to a reactor trip and therefore used Table 3.1, “SDP Worksheet for Fermi 2 Nuclear Power Plant – Transients (Reactor Trip)(TRANS).” Because the shield block walls were removed for less than three days, the inspectors assigned an initiating event likelihood of 3. The affected safety functions were the power conversion system and depressurization; however, full mitigation credit for depressurization remained because this finding did not affect the safety relief valves. The most limiting cutset was the transient-power conversion system-containment heat removal-containment venting with a result of 8 without any credit for operator recovery. The inspectors determined that this finding did not affect either the containment heat removal or the containment venting and therefore assigned full creditable mitigation capability for each of those two safety functions. Using the counting rule worksheet with 1 sequence with a risk significance of 9 and 1 sequence with a risk significance of 8 yielded an overall result of very low safety significance (Green).

The licensee entered this issue into their corrective action program as CARD 07-25277. Immediate corrective actions included hourly steam tunnel temperature monitoring and the erection of a temporary wall to restore normal building airflow. The inspectors determined the finding was associated with a cross-cutting aspect in the area of Human Performance, Work Control, because the licensee failed to appropriately plan the work activity by incorporating risk insights (H.3(a)).

Enforcement: 10 CFR 50.65(a)(4) requires, in part, that the licensee assess and manage the increase in risk that may result from a maintenance activity prior to performing the activity. Contrary to the above, on September 19, 2007, the licensee failed to adequately assess the increase in risk in removing a shield block wall. The failure to adequately assess risk prior to performing a maintenance activity is a violation of 10 CFR 50.65(a)(4) in accordance with paragraph 7.11.1.d.1(b)(4) of the NRC Enforcement Manual. Because this finding was of very low safety significance and because it was entered into the licensee’s corrective action program as CARD 07-25277, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000341/2007006-04, Inadequate Work Planning for Shield Block Wall Removal.

1R15 Operability Evaluations (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- CARD 07-26412, Evaluate RHR Reservoir Emergency Pump Column Inspection Results;
- CARD 07-26464, RCIC Steam Isolation Valve Heat Stressed Cables;
- CARD 07-26974, Plastic Ty-Wraps in Drywell; and
- CARD 07-27395, B1 Reactor Protection System (RPS) Trip String Failure to Actuate on Simulated Level 2.

The inspectors selected these potential operability issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TS and UFSAR to the licensee's evaluations to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations.

These activities constitute four operability evaluations samples as defined in Inspection Procedure 71111.15-05.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

.1 Annual Resident Inspector Review

a. Inspection Scope

The following EDP was reviewed and selected aspects were discussed with engineering personnel:

- EDP-33703, Division II Emergency Equipment Cooling Water (EECW) Pump Replacement Modification.

This document and related documentation were reviewed for adequacy of the associated 10 CFR 50.59 safety evaluation screening, consideration of design parameters, implementation of the modification, post-modification testing, and relevant procedures, design, and licensing documents were properly updated. The inspectors observed ongoing and completed work activities to verify installation was consistent with the design control documents. The modification replaced the Division II pump and motor set with a new unit.

These inspection activities constitute one annual sample as defined in Inspection Procedure 71111.17-05.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the following post-maintenance testing (PMT) activities to verify procedures and test activities were adequate to ensure system operability and functional capability:

- Work Order (WO) A002961018 and WO 2482909, Containment Area High Range Radiation Monitor Calibration;
- WO 25129754, Replace EECW Division I Pump;
- WO 00Z053614, Refurbish Actuator for System Valve E1150F028B;
- WO 0984070929, Reactor Pressure Vessel Hydrostatic Test Following Control Rod Drive Mechanism Replacements;
- WO 000Z033471, Replace Reactor Water Cleanup Check Valve G33F120;
- WO 000Z052900, Replace RHR Pump Motor 'A';
- WO 1138030328, Drywell Integrated Leak Rate Test Following Drywell Cooler Replacements;
- WO D7830990100, Replace MSIV 'D' Actuator Springs;
- WO E520961116, RCIC Pump 10-Year ISI Inspection; and
- Number 5 Low Pressure Stop Valve (LPSV) Surveillance 23.109 for Post-Maintenance Testing (PMT).

These activities were selected based upon the structure, system, or component's ability to impact risk. The inspectors evaluated these activities for the following (as applicable): the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was returned to its operational status following testing (temporary modifications or jumpers required for test performance were properly removed after test completion), and test documentation was properly evaluated. The inspectors evaluated the activities against TS, the UFSAR, 10 CFR 50 requirements, licensee procedures, and various NRC generic communications to ensure the test results adequately ensured the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with PMT to determine whether the licensee was identifying problems and entering them in the corrective action program and that the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the Attachment.

These inspection activities constitute 10 PMT samples as defined in Inspection Procedure 71111.19.

b. Findings

(1) Failure to Maintain Configuration Control of the RCIC Pump

Introduction: A self-revealed Green NCV of 10 CFR 50, Appendix B, Criterion III, "Design Control," was identified for the failure to properly maintain design control of the RCIC pump.

Description: On November 14, during plant startup from RF12, the licensee performed a RCIC pump functional test at 150 psig reactor pressure and declared RCIC operable following successful completion of the test. On November 17, the licensee performed the RCIC pump and valve operability test at 950 psig reactor pressure and declared the pump inoperable when it failed to meet acceptance criteria for total dynamic head. The licensee formed an emergent issues team to determine the cause of the test failure. The licensee determined the cause of the low total dynamic head was likely due to inadequate parts that were installed during RF12.

During RF12, the licensee performed a ten-year overhaul of the RCIC pump. Prior to the outage, the licensee procured a replacement rotating assembly for the pump in accordance with the information contained from approved design documents. Specifically, those documents identified the diameters for all four impellers and that information was used to specify the replacement impeller diameters.

During the troubleshooting following the test failure, the licensee compared the old rotating assembly to the new rotating assembly and found discrepancies in the impeller diameters. The third stage impeller in the old rotating assembly was larger than the same stage impeller for the new rotating assembly. The licensee concluded that the undersized impeller and a change in the casting process for the old and new impellers were the causes of the 950 psig test failure on November 17. The undersized impeller, however, was sufficient to prevent the pump from fulfilling its intended safety function.

The inspectors reviewed documentation surrounding the last pump overhaul performed in 1996 and determined the licensee did not verify the impeller dimensions prior to their replacement in 1996. The inspectors also learned that the licensee sent impellers from their stock to the pump manufacturer so the pump manufacturer could assemble a complete rotating assembly; however, the licensee had no receipt inspection documents for those impellers to identify they were the correct diameter. Consequently, neither the inspectors nor the licensee could conclusively determine how the two larger impellers came to be installed in the RCIC pump during the 1996 pump overhaul without the relevant design documents being modified to show the actual diameters.

The licensee entered this issue into their corrective action program as CARD 07-27462. The licensee continued to investigate the issue and form their long-term corrective actions. As a result of the test failure, the licensee re-installed the old rotating assembly under EDP 35464 and updated the relevant design documents to reflect the actual impeller diameters currently installed in the RCIC pump.

Analysis: The inspectors determined the failure to maintain adequate configuration control for the RCIC pump was a performance deficiency because the RCIC pump was safety-related and the licensee was expected to comply with 10 CFR 50, Appendix B, Criterion III. The inspectors determined the finding was more than minor in accordance with IMC 0612, Appendix B, Section 3, dated September 30, 2005. The inspectors compared this issue with the examples contained in Appendix E of IMC 0612 and determined that example 3.b was related to this issue. From this example, the inspectors concluded the issue was more than minor because the incorrect design documents were used to procure an inadequately-sized impeller; the impeller was later installed in the pump; the undersized impeller adversely affected the operation of the pump; and the pump was returned to service prior to discovery of the problem.

The inspectors determined this finding affected the Mitigating Systems Cornerstone. This finding was of very low safety significance because the finding did not result in a loss of a system safety function because High Pressure Coolant Injection remained available during all relevant time periods post-RF12, did not represent an actual loss of safety function for RCIC for greater than its technical specification allowed outage time and did not screen as potentially risk significant from external events. Immediate corrective actions included reinstalling the old rotating assembly and updating the design documents to reflect the as-built RCIC configuration.

Enforcement: 10 CFR 50, Appendix B, Criterion III, requires, in part, that the licensee correctly translate the as-built design configuration of safety-related equipment into the appropriate design documents. The RCIC pump is and always has been classified as safety-related in the licensee's quality assurance program. Contrary to the above, on October 18, 1996, the licensee installed a new rotating assembly in the RCIC pump and failed to update the design documents to reflect the as-built impeller diameters. Because this finding was of very low safety significance and because it was entered into the licensee's corrective action program as CARD 07-27462, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000341/2007006-05, Failure to Maintain Configuration Control of the RCIC Pump.

(2) Reactor Power and Water Level Transient During Testing of Turbine LPSV #5

Introduction: A Green self-revealed finding was identified for the failure to operate the plant in accordance with the documented instructions during a maintenance activity on the number 5 LPSV. No violation of regulatory requirements was identified.

Description: On November 17, 2007, the inspectors reviewed the unplanned power transient caused by the inadvertent cycling the main turbine bypass valves. Following maintenance on the number 5 LPSV position indications, maintenance requested operators to stroke number 5 LPSV for the PMT. Operators reviewed Procedure 23.109, "Turbine Operating Procedure," section 5.1, and noted that according to the procedure, some control room alarms needed to be cleared. However, based on input from a maintenance supervisor, operators continued with the PMT. Procedure 23.109 required the operator to turn the governor interlock switch to the exercise permit position but instead, the operator inadvertently turned it to the governor interlock position. The mis-positioning of the switch caused the turbine bypass valves to stroke open. Upon realizing the error, the operator released the switch and the bypass valves cycled full closed.

As a result of the inadvertent bypass valve cycling, the maximum reactor power transient was an increase of 7.7 percent. Additionally, reactor water level increased to within 3 inches of the reactor water level 8 setpoint and reactor pressure decreased from 947 psig to 913 psig. All parameters later returned to their pre-transient valves and no safety-related actuations or isolations occurred as a result of this event.

Analysis: The inspectors determined the failure to properly operate the plant in accordance with the documented instructions while performing a maintenance activity was a performance deficiency because licensee personnel are expected to comply with procedures. The inspectors determined the finding was more than minor in accordance with IMC 0612, Appendix B, Section 3, dated September 30, 2005. The inspectors compared this issue with the examples contained in Appendix E and determined that example 4.b was related to this issue because the operators' inadvertent operation of the governor interlock switch caused an unplanned plant transient. The inspectors determined this finding affected the Initiating Events Cornerstone and was of very low safety significance because although a power transient occurred, it had no affect on the availability of mitigation equipment or functions. The licensee entered this issue into their corrective action program as CARD 07-27478. Immediate corrective actions included stabilizing the plant, reviewing the plant response to the transient to ensure all systems operated as designed, and re-emphasizing the importance and expectation that the operations department would follow plant procedural use and adherence policy. The inspectors determined the finding was associated with a cross-cutting aspect in the area of Human Performance, Work Practices, because licensee personnel failed to follow written procedures when manipulating plant equipment (H.4(b)).

Enforcement: No violation of regulatory requirements was identified because the main turbine low pressure stop and bypass valves were not classifies as safety-related in the licensee's quality assurance program. This issue was entered into the licensee's corrective action program as CARD 07-27478. This finding is described as FIN 05000341/2007006-06: Transient During Testing of Number 5 Low Pressure Stop Valve.

1R20 Outage Activities (71111.20)

.1 Refueling Outage Activities

a. Inspection Scope

The inspectors reviewed the outage risk and contingency plans for the Unit 2 refueling outage conducted September 29 through November 18, 2007, to confirm the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below:

- licensee configuration management, including maintenance of defense-in-depth commensurate with the risk plan for key safety functions and compliance with the applicable TS when taking equipment out-of-service;

- implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing;
- installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication and an accounting for instrument error;
- controls over the status and configuration of electrical systems to ensure TS and outage safety plan requirements were met, and controls over switchyard activities;
- monitoring of decay heat removal processes;
- controls to ensure outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system;
- reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss;
- controls over activities that could affect reactivity;
- maintenance of secondary containment as required by TS;
- refueling activities, including fuel handling and sipping to detect fuel assembly leakage;
- startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the drywell (primary containment) to verify debris had not been left which could block ECCS suction strainers, and reactor physics testing; and
- licensee identification and resolution of problems related to refueling outage activities.

Documents reviewed during the inspection are listed in the Attachment.

These inspection activities constitute one refueling outage sample as defined in Inspection Procedure 71111.20-05.

b. Findings

No findings of significance were identified.

.2 Non-Qualified Ty-Wraps Inside Primary Containment

a. Inspection Scope

During the refueling outage, the inspectors performed walkdown inspections inside the drywell looking for debris that could migrate into the ECCS suction strainers. The inspectors noted there were numerous ty-wraps and shield blankets in the drywell and pealing paint in the torus. The inspectors questioned the licensee about the controls for potential debris in the torus and the effects that the debris could have on the performance of ECCS equipment. The inspectors reviewed the licensee’s controls against previously-performed design calculations and other engineering evaluations to ensure that the licensee was in compliance with the current analyses.

b. Findings and Observations

Introduction: The inspectors identified an unresolved item related to the uncontrolled use of ty-wraps in the drywell.

Description: The inspectors noted the use of Tefzel ty-wraps inside primary containment and, due to their abundance, the inspectors questioned the licensee about their tracking program for items that could become debris in the torus and potentially block the ECCS and RCIC pump suctions. The licensee stated the design specification did not cover the use of ty-wraps in the drywell. Upon further investigation, the licensee identified that the lead shielding installed during a previous outage had not been considered in the design specification. The inspectors reviewed specification 3071-389, Revision 3, "Emergency Core Cooling System Suction Strainers," and noted that ty-wraps were not addressed in the specification. When questioned by the inspectors, the licensee stated the only program in place was to track paint in the drywell and torus. Upon further review, the licensee noted shield blankets had been installed in the drywell without modeling the debris loading on the strainers.

The licensee-performed walkdowns revealed a total of 36 ft^2 of ty-wraps inside the drywell. The licensee removed 20.6 ft^2 of ty-wraps from the drywell following the walkdowns. Engineering personnel performed Engineering Functional Analysis (EFA) E11-07-0005, Revision A, "Non-Qualified, Tefzel Ty-Wraps Inside Primary Containment," to analyze the effects of the remaining ty-wraps and the shield blanket debris on ECCS and RCIC suction from the torus. This analysis included the potential for the ty-wraps to be carried into the strainers via the torus for several different accident scenarios.

The strainer hydraulic analysis performed in the EFA allowed for an increase in the load from fibrous debris and included the lead wool and ty-wraps. The evaluation concluded that only 7.1 ft^2 of ty-wraps were available for transport to the strainers, which was less than the maximum allowable load of 13.75 ft^2. For the design limiting case, the RHR and Core Spray pumps would have sufficient net positive suction head available thereby alleviating current safety concerns. Resolution of the past operability concern will be provided by incorporating the remaining material in the drywell into the plant design bases associated with strainer loading and the establishment of a program for control of materials inside containment.

Pending further analysis of past operability by the licensee of whether the use of non-qualified Tefzel ty-wraps inside primary containment could have plugged the strainers, this is considered an Unresolved Item. URI 05000341/2007006-07: Non-Qualified Ty-Wraps Inside Primary Containment.

1R22 Surveillance Testing (71111.22)

.1 Routine Surveillance Testing

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- Feedwater Check Valve Local Leak Rate Testing (LLRT);
- Final Core Load Verification;
- Spent Fuel Pool Loading; and
- Loss of Offsite Power/Loss of Coolant Accident Testing.

The inspectors observed in-plant activities and reviewed procedures and associated records to determine whether: preconditioning occurred; effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing; acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis; plant equipment calibration was correct, accurate, and properly documented; as-left setpoints were within required ranges; the calibration frequency was in accordance with TSs, the UFSAR, procedures, and applicable commitments; measuring and test equipment calibration was current; test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied; test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used; test data and results were accurate, complete, within limits, and valid; test equipment was removed after testing; where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable; where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure; where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished; prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test; equipment was returned to a position or status required to support the performance of its safety functions; and all problems identified during the testing were appropriately documented and dispositioned in the corrective action program. Documents reviewed are listed in the Attachment.

These inspection activities constitute four routine surveillance testing samples as defined in Inspection Procedure 71111.22.

b. Findings

No findings of significance were identified.

.2 In-service Testing

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- RHR Discharge Check Valve LLRT;
- Division II EECW Pump Test Following Pump Replacement; and
- Division I Low Pressure Coolant Injection and Torus Cooling/Spray Pump and Valve Operability Test.

The inspectors observed in-plant activities and reviewed procedures and associated records to determine whether: preconditioning occurred; effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing; acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis; plant equipment calibration was correct, accurate, and properly documented; as-left setpoints were within required ranges; the calibration frequency was in accordance with TSs, the UFSAR, procedures, and applicable commitments; measuring and test equipment calibration was current; test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied; test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used; test data and results were accurate, complete, within limits, and valid; test equipment was removed after testing; where applicable for in-service testing activities, testing was performed in accordance with the applicable version of Section XI, ASME Code, and reference values were consistent with the system design basis; where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable; where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure; where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished; prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test; equipment was returned to a position or status required to support the performance of its safety functions; and all problems identified during the testing were appropriately documented and dispositioned in the corrective action program. Documents reviewed are listed in the Attachment.

These inspection activities constitute three in-service inspection samples as defined in Inspection Procedure 71111.22.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

.1 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the following temporary modifications:

- Outage Electrical Temporary Modifications for Fire Protection and the Control Room; and
- Temporary Modification 07-0026, On-Line Leak Repair of Steam Leak on N30-F006 .

The inspectors compared the temporary configuration changes and associated 10 CFR 50.59 screening and evaluation information against the design basis, the UFSAR, and the TSs, as applicable, to verify the modifications did not affect the operability or availability of the affected systems. The inspectors also compared the licensee’s information to operating experience information to ensure lessons learned from other utilities had been incorporated into the licensee’s decision to implement the temporary modifications. The inspectors, as applicable, performed field verifications to ensure the modifications were installed as directed; the modifications operated as expected; modification testing adequately demonstrated continued system operability, availability, and reliability; and operation of the modifications did not impact the operability of any interfacing systems. Lastly, the inspectors discussed the temporary modification with operations, engineering, and maintenance personnel to ensure the individuals were aware of how operation with the temporary modification in place could impact overall plant performance.

These inspection activities constitute two temporary plant modifications samples as defined in Inspection Procedure 71111.23-05.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspectors performed a screening review of Revisions 32 and 33 of the Fermi 2 Power Plant Emergency Plan to determine whether changes identified in Revisions 32 and 33 decreased the effectiveness of the licensee's emergency planning for the Fermi Power Plant. This review did not constitute an approval of the changes, and as such, the changes are subject to future NRC inspection to ensure the emergency plan continues to meet NRC regulations.

These inspection activities constitute one inspection sample.

b. Findings

No findings of significance were identified.

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Plant Walkdowns and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors reviewed licensee controls and surveys in the following radiologically significant work areas within radiation areas, high radiation areas (HRA), and airborne radioactivity areas in the plant and reviewed work packages which included associated licensee controls and surveys of these areas to determine if radiological controls including surveys, postings, and barricades were acceptable:

- Reactor Vessel Head Cleaning; and

- Replacement of the South RWCU Pump Motor.

These inspection activities constitute one sample as defined by Inspection Procedure 71121.01-5.

The identified radiologically significant work areas were walked down and surveyed to evaluate whether the prescribed radiation work permit (RWP), procedures and engineering controls were in place, licensee surveys and postings were complete and accurate, and air samplers were properly located.

These inspection activities constitute one inspection sample.

The inspectors reviewed selected RWPs and associated radiological controls used to access these and other radiologically significant areas and evaluated the work control instructions and control barriers that were specified in order to evaluate whether the controls and requirements provided adequate worker protection. Site TS requirements for HRAs and locked HRAs were used as standards for the necessary barriers. Electronic dosimeter alarm set points for both integrated dose and dose rate were evaluated for conformity with survey indications and plant policy. The inspectors attended pre-job briefings to assess whether instructions to workers emphasized the actions required when their electronic dosimeters noticeably malfunctioned or alarmed.

These inspection activities constitute one inspection sample.

.2 Job-In-Progress Reviews

a. Inspection Scope

The inspectors observed the following jobs that were being performed in radiation areas and HRAs for observation of work activities that presented the greatest radiological risk to workers:

- Reactor Vessel Head Cleaning; and
- Replacement of the South RWCU Pump Motor.

The inspectors reviewed radiological job requirements for these activities, including RWP requirements and work procedure requirements, and attended as-low-as-reasonably-achievable (ALARA) job briefings.

These inspection activities constitute one inspection sample.

Job performance was observed with respect to these requirements to assess whether radiological conditions in the work area were adequately communicated to workers through pre-job briefings and postings. The inspectors also evaluated the adequacy of radiological controls including required radiation, contamination, and airborne surveys for system breaches; radiation protection job coverage which included audio and visual surveillance for remote job coverage; and contamination controls.

These inspection activities constitute one inspection sample.

The inspectors reviewed the adequacy of radiological controls, radiation protection job coverage (including audio and visual surveillance for remote job coverage), and contamination controls during these job performance observations.

These inspection activities constitute one inspection sample.

The inspectors reviewed the application of dosimetry to effectively monitor exposure to personnel for high radiation work areas with significant dose rate gradients (factor of 5 or more).

These inspection activities constitute one inspection sample.

b. Findings

No findings of significance were identified.

.3 High Radiation, Locked High Radiation Area and Very High Radiation Area Access Controls

a. Inspection Scope

The inspectors reviewed the licensee's procedures and radiation protection (RP) job standards and evaluated RP practices for the control of access to radiologically significant areas (high, locked high, and very high radiation areas). The inspectors discussed locked high and very high radiation area controls with the RP staff to assess compliance with the licensee's TS, procedures, and the requirements of 10 CFR 20 and for consistency with the guidance contained in Regulatory Guide 8.38.

b. Findings

Introduction: A Green self-revealed finding of very low safety significance and an associated NCV of TS 5.7.1.b was identified for workers entering an HRA without an adequate awareness of radiological conditions and while working under an RWP that did not allow entry into the area.

Description: On October 6, 2007, two supplemental radiation workers received a briefing from an RP technician and entered the torus basement area to locate a snubber. The electronic dosimeter dose rate alarm setpoint for the two workers was established at 90 millirem/hour. The workers exited the area and learned that a dose rate alarm occurred through the access control computer system. The workers stated that the alarm was not heard.

During the RP briefing, the two workers asked a radiation protection technician (RPT) whether they could enter an area in the torus basement they described as a contaminated area to search for a snubber. The RPT authorized the activity and briefed the workers of the radiological conditions of the contaminated area he thought would be entered. After the briefing, the workers left the RPT to continue their search for the snubber. The workers then entered an area posted as an HRA and contaminated area, an area that was not authorized by the RWP or by the RPT. The workers believed they were authorized to enter the area based on the briefing from the RPT. During that briefing, the RPT did not discuss HRA requirements nor did the RPT instruct the workers not to enter an HRA. Consequently, the radiation workers entered the HRA without being aware of the radiological conditions of the area and on an RWP that did not allow access to HRAs.

Analysis: Entering an HRA without being aware of the radiological conditions and without being on an RWP that allowed HRA access as required by the licensee's TS 5.7.1.b represents a performance deficiency as defined in NRC IMC 0612, Appendix B, "Issue Screening." The inspectors determined the issue was associated with the Program/Process attribute of the Occupational Radiation Safety Cornerstone and affected the cornerstone objective to ensure adequate protection of worker health and safety from exposure to radiation. Therefore, the issue was more than minor and represented a finding which was evaluated using the SDP.

Since the finding involved a radiological access control problem, the inspectors utilized IMC 0609, Appendix C, "Occupational Radiation Safety SDP," to assess its significance. The inspectors determined this finding did not involve: (1) an ALARA finding; (2) an overexposure; (3) a substantial potential for overexposure; or (4) an impaired ability to assess doses. Consequently, the inspectors concluded the SDP assessment for this finding was of very low safety significance (Green). Additionally, this finding has a cross-cutting aspect in the area of Human Performance because the RPT did not validate the intended work area and the radiation workers did not perform a self check or peer check of the requirements needed before entering the HRA (H.4(a)).

Enforcement: TS 5.7.1.b requires that personnel be aware of radiological conditions of HRAs before entry and that entries be controlled using an RWP. Contrary to this requirement, personnel entered an HRA in the torus basement without the required knowledge of radiological conditions and without an RWP that allowed access to an HRA.

Corrective actions taken by the licensee included performance management (coaching) of the involved personnel. The licensee also performed additional communications to the plant population through flyers and a site stand-down to reinforce that all radiation workers ensure they read all posted signs and work on the correct RWP. Since the licensee documented this issue in its corrective action program (CARD 07-25796) and because the violation is of very low safety significance, it is being treated as NCV 05000341/2007006-08, HRA Entry Without Adequate Awareness of Radiological Conditions and Without an RWP that Allowed HRA Entry.

2OS2 As-Low-As-Is-Reasonably-Achievable Planning and Controls (71121.02)

.1 Radiological Work Planning

a. Inspection Scope

The interfaces between operations, RP, maintenance, maintenance planning, scheduling, and engineering groups, were evaluated to identify any interface problems or missing program elements.

These inspection activities constitute one inspection sample.

The integration of ALARA requirements into work procedures and RWP documents was evaluated to assess whether the licensee's radiological job planning would reduce dose.

These inspection activities constitute one inspection sample.

Shielding requests from the radiation protection group were evaluated with respect to dose rate reduction and reduced worker exposure.

These inspection activities constitute one inspection sample.

The inspectors reviewed work activity planning to determine if there was consideration of the benefits of dose rate reduction activities such as shielding provided by water filled components and piping, job scheduling, along with shielding and scaffolding installation and removal activities.

These inspection activities constitute one inspection sample.

b. Findings

No findings of significance were identified.

.2 Verification of Dose Estimates and Exposure Tracking Systems

a. Inspection Scope

The inspectors reviewed the assumptions and basis for the current annual collective exposure estimate. The inspectors reviewed applicable procedures to evaluate the methodology for estimating work activity-specific exposures and the intended dose outcome. The inspectors evaluated both dose-rate and man-hour estimates for reasonable accuracy.

These inspection activities constitute one inspection sample.

The inspectors reviewed the licensee's exposure tracking system. The inspectors assessed whether the level of exposure tracking detail, exposure report timeliness and exposure report distribution was sufficient to support control of collective exposures.

These inspection activities constitute one inspection sample.

b. Findings

No findings of significance were identified.

.3 Source-Term Reduction and Control

a. Inspection Scope

The inspectors discussed with the licensee its plans to perform a comprehensive source-term evaluation including the input mechanisms in order to reduce the source-term. The licensee indicated the evaluation would prescribe a new source-term control strategy that may include a process for evaluating radionuclide distribution plus a shutdown and operating chemistry plan, which can minimize the source-term external to the core. Other methods used by the licensee to control the source-term, including component/system decontamination, hotspot flushing, and the use of shielding, were evaluated.

These inspection activities constitute one inspection sample.

The licensee's process for identification of specific sources was reviewed, along with exposure reduction actions and the priorities the licensee had established for implementation of those actions. Results achieved against these priorities since the last refueling cycle were reviewed. For the current assessment period, source-term reduction evaluations were reviewed, and actions taken to reduce the overall source-term were compared to the previous year.

These inspection activities constitute one inspection sample.

b. Findings

No findings of significance were identified.

.4 Monitoring of Declared Pregnant Women and Dose to Embryo/Fetus

a. Inspection Scope

The inspectors reviewed the licensee's monitoring methods and procedures, radiation exposure controls, and the information provided to declared pregnant women to determine if an adequate program had been implemented to limit embryo/fetal dose. The inspectors also reviewed the pregnancy declaration forms and the radiation exposure information for several individuals who declared their pregnancy to the licensee between 2006 and 2007 to determine if the licensee met the requirements of 10 CFR 20.1208 and 20.2106.

These inspection activities constitute one inspection sample.

b. Findings

No findings of significance were identified.

.5 Problem Identification and Resolutions

a. Inspection Scope

Portions of the licensee's corrective action and self-assessment programs were reviewed to determine if repetitive deficiencies and/or significant individual deficiencies in problem identification and resolution had been addressed.

These inspection activities constitute one inspection sample.

b. Findings

No findings of significance were identified.

2PS2 Radioactive Material Processing and Transportation (71122.02)

.1 Radioactive Waste System

a. Inspection Scope

The inspectors reviewed the liquid and solid radioactive waste system description in the UFSAR for information on the types and amounts of radioactive waste generated and disposed. The inspectors reviewed the scope of the licensee's audit program with regard to radioactive material processing and transportation programs to assess whether it met the requirements of 10 CFR 20.1101(c).

These inspection activities constitute one sample as defined by Inspection Procedure 71122.02-5.

b. Findings

No findings of significance were identified.

.2 Radioactive Waste System Walkdowns

a. Inspection Scope

The inspectors performed walkdowns of the liquid and solid radioactive waste processing systems to assess whether the systems agreed with the descriptions in the UFSAR and the Process Control Program and to assess the material condition and operability of the systems. The inspectors reviewed the status of radioactive waste process equipment that was not operational and/or was abandoned in place. The inspectors reviewed the licensee's administrative and physical controls to ensure the equipment would not contribute to an unmonitored release path or be a source of unnecessary personnel exposure.

The inspectors reviewed changes to the waste processing system to assess whether the changes were reviewed and documented in accordance with 10 CFR 50.59 and to assess the impact of the changes on radiation dose to members of the public. The inspectors reviewed the current processes for transferring waste resin into shipping containers to evaluate whether the appropriate waste stream mixing and/or sampling procedures were utilized. The inspectors also reviewed the methodologies for waste concentration averaging to evaluate if representative samples of the waste product were provided for the purposes of waste classification in 10 CFR 61.55.

These inspection activities constitute one sample as defined in Inspection Procedure 71122.02-5.

b. Findings

No findings of significance were identified.

.3 Waste Characterization and Classification

a. Inspection Scope

The inspectors reviewed the licensee's radiochemical sample analysis results for each of the licensee's waste streams, including dry active waste, spent resins and filters. The inspectors also reviewed the licensee's use of scaling factors to quantify difficult-to-measure radionuclides (e.g., pure alpha or beta emitting radionuclides). The reviews were conducted to assess whether the licensee's program assured compliance with 10 CFR 61.55 and 10 CFR 61.56, as required by Appendix G of 10 CFR 20. The inspectors also reviewed the licensee's waste characterization and classification program to ensure the waste stream composition data accounted for changing operational parameters and thus remained valid between the annual sample analysis updates.

These inspection activities constitute one sample as defined in Inspection Procedure 71122.02-5.

b. Findings

No findings of significance were identified.

.4 Shipment Preparation

a. Inspection Scope

The inspectors reviewed shipment packaging, surveying, labeling, marking, placarding, vehicle checks, emergency instructions, disposal manifest, shipping papers provided to the driver, and licensee verification of shipment readiness for a contaminated laundry shipment. The inspectors assessed whether the requirements of any applicable transport cask Certificate of Compliance were met and assessed whether the receiving licensee was authorized to receive the shipment packages. The inspectors assessed whether the licensee's procedures for cask loading and closure procedures were consistent with the vendor's approved procedures. The inspectors observed radiation worker practices to assess whether the workers had adequate skills to accomplish each task and to assess whether the shippers were knowledgeable of the shipping regulations and whether shipping personnel demonstrate adequate skills to accomplish the package preparation requirements for public transport with respect to NRC Bulletin 79-19 and 49 CFR 172 Subpart H. The inspectors reviewed the training provided to personnel responsible for the conduct of radioactive waste processing and radioactive shipment preparation activities. The review was conducted to assess whether the licensee's training program provided training consistent with NRC and Department of Transportation requirements.

These inspection activities constitute one sample as defined in Inspection Procedure 71122.02-5.

b. Findings

No findings of significance were identified.

.5 Shipping Records

a. Inspection Scope

The inspectors reviewed six non-excepted package shipment manifests/documents completed in 2005 through 2007 to assess compliance with NRC and Department of Transportation requirements (i.e., 10 CFR Parts 20 and 71 and 49 CFR Parts 172 and 173).

These inspection activities constitute one sample as defined in Inspection Procedure 71122.02-5.

b. Findings

No findings of significance were identified.

.6 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed audits and self assessments that addressed radioactive waste and radioactive materials shipping program deficiencies since the last inspection to assess whether the licensee had effectively implemented the corrective action program and that problems were identified, characterized, prioritized and corrected. The inspectors also assessed whether the licensee's self-assessment program was capable of identifying repetitive deficiencies or significant individual deficiencies in problem identification and resolution.

The inspectors also reviewed corrective action reports from the radioactive material and shipping programs since the previous inspection, interviewed staff and reviewed documents to determine if the following activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk:

- initial problem identification, characterization, and tracking;
- disposition of operability/reportability issues;
- evaluation of safety significance/risk and priority for resolution;
- identification of repetitive problems;
- identification of contributing causes;
- identification and implementation of effective corrective actions;
- resolution of NCVs tracked in corrective action system(s); and
- implementation/consideration of risk significant operational experience feedback.

These inspection activities constitute one sample as defined in Inspection Procedure 71122.02-5.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Review of Licensee's Quarterly PI Data Submittal

a. Inspection Scope

The inspectors performed a review of the data submitted by the licensee for the fourth quarter 2007 performance indicators for any obvious inconsistencies prior to its public release in accordance with IMC 0608, "Performance Indicator Program."

This review was performed as part of the inspectors' normal plant status activities and, as such, did not constitute a separate inspection sample.

b. Findings

No findings of significance were identified.

.2 Safety System Functional Failures

a. Inspection Scope

The inspectors sampled licensee submittals for the Safety System Functional Failures performance indicator (PI) for the period from the first quarter 2006 through the third quarter 2007. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in Revision 5 of the Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," and NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73," definitions and guidance were used. The inspectors reviewed the licensee's operator narrative logs, operability assessments, maintenance rule records, maintenance work orders, issue reports, event reports, and NRC integrated inspection reports for the first quarter 2006 through the third quarter 2007 to validate the accuracy of the submittals. Specific documents reviewed are described in the Attachment.

These inspection activities constitute one safety system functional failures sample as defined in Inspection Procedure 71151.

b. Findings

No findings of significance were identified.

.3 Mitigating Systems Performance Index (MSPI) - Emergency AC Power System

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI - Emergency AC Power System PI for the period from the first quarter 2007 through the third quarter 2007. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in Revision 5 of the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," were used. The inspectors reviewed the licensee's operator narrative logs, MSPI derivation reports, CARDs, event reports, and NRC integrated inspection reports for the period of the first quarter 2007 through the third quarter 2007 to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection and, if so, that the change was in accordance with applicable NEI guidance. Specific documents reviewed are described in the Attachment.

These inspection activities constitute one MSPI emergency AC power system sample as defined in Inspection Procedure 71151.

b. Findings

No findings of significance were identified.

.4 Mitigating Systems Performance Index - Cooling Water Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI - Cooling Water Systems PI for the period from the first quarter 2007 through the third quarter 2007. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in NEI Document 99-02 were used. The inspectors reviewed the licensee's operator narrative logs, CARDs, MSPI derivation reports, event reports, and NRC integrated inspection reports for the period of the first quarter 2007 through the third quarter 2007 to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. Specific documents reviewed are described in the Attachment.

These inspection activities constitute one MSPI cooling water system sample as defined in Inspection Procedure 71151.

b. Findings

No findings of significance were identified.

.5 Reactor Coolant System (RCS) Specific Activity

a. Inspection Scope

The inspectors sampled licensee submittals for the RCS Specific Activity PI for the period from the first quarter 2007 through the third quarter 2007. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in NEI Document 99-02 were used. The inspectors reviewed the licensee's RCS chemistry samples, TS requirements, CARDs, event reports, and NRC integrated inspection reports for the period of the first quarter 2007 through the third quarter 2007 to validate the accuracy of the submittals. Specific documents reviewed are described in the Attachment.

These inspection activities constitute one reactor coolant system specific activity sample as defined in Inspection Procedure 71151.

b. Findings

No findings of significance were identified.

.6 Radiological Effluent TS/Offsite Dose Calculation Manual (ODCM) Radiological Effluent Occurrences

a. Inspection Scope

The inspectors sampled licensee submittals for the Radiological Effluent TS ODCM Radiological Effluent Occurrences performance indicator for the period from the third quarter 2006 through the fourth quarter 2007 to evaluate the accuracy of the PI data reported during those periods. The PI definitions and guidance contained in NEI Document 99-02 were used. The inspectors reviewed the licensee's CARD database and selected individual reports generated since this indicator was last reviewed to identify any potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose.

These inspection activities constitute one Radiological Effluent TS/ODCM radiological effluent occurrences sample as defined in Inspection Procedure 71151.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution (71152)

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, and Occupational Radiation Safety
Routine Review of Items Entered Into the Corrective Action Program

a. Inspection Scope

As part of the various baseline inspection procedures discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify they were being entered into the licensee's corrective action program at an appropriate threshold, adequate attention was being given to timely corrective actions, and adverse trends were identified and addressed. Attributes reviewed included: the complete and accurate identification of the problem; that timeliness was commensurate with the safety significance; that evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent-of-condition reviews, and previous occurrences reviews were proper and adequate; and that the classification, prioritization, focus, and timeliness of corrective actions were commensurate with safety and sufficient to prevent recurrence of the issue.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead, by procedure they were considered an integral part of the inspections performed during the quarter and documented in this report.

b. Findings

No findings of significance were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

In order to assist with the identification of repetitive equipment failures and specific Human Performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. This review was accomplished through inspection of the station's daily condition report packages.

These daily reviews were performed by procedure as part of the inspectors' daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. Findings

No findings of significance were identified.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's corrective action program and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues but also considered the results of daily inspector corrective action item screening discussed in Section 4OA2.2 above, licensee trending efforts, and licensee Human Performance results. The inspectors' review nominally considered the six-month period of July 1, 2007, through December 31, 2007, although some examples expanded beyond those dates where the scope of the trend warranted.

The review also included issues documented outside the normal corrective action program in system health reports, quality assurance audit/surveillance reports, and self-assessment reports. The inspectors compared and contrasted their results with the results contained in the licensee's corrective action program trending reports. Corrective actions associated with a sample of the issues identified in the licensee's trending reports were reviewed for adequacy.

These inspection activities constitute one semi-annual trend inspection sample.

b. Issues

The inspectors noted an increase in the number of Human Performance-related issues during the fourth quarter. The number of Human Performance clock resets in maintenance during the refueling outage was 17; this is a significant increase over the last several outages. The inspectors noted three findings in this inspection period associated with Human Performance cross-cutting issues were attributed to multiple departments. In addition, the inspectors noted the Human Performance issues increase was related to several significant issues. During the current semi-annual trend review, the inspectors noted more examples of Human Performance issues:

- a hole drilled in the safety relief valve tailpipe that was evaluated in report 2007-010;
- loss of radio communications in the control room during the loss-of-power testing;
- reactor transient caused by inadvertently opening and closing the turbine bypass valves;
- removal of turbine shield blocks that caused an unexpected increase in turbine building temperature; and
- low-level scram as a result of a reference leg perturbation caused by improper operation of a valve.

The licensee already identified this trend and has begun developing a plan to address the causes. The inspectors also noted the licensee has identified a potential emerging trend in ineffective/inadequate corrective actions (CARDs 07-27943, 07-28135, and 07-298134). This trend is corroborated by the increased number of problem identification and resolution cross-cutting issues during the last 2 quarters. The inspectors noted that the licensee had ten level 1 CARDs, each with an associated root cause evaluation, during the previous two quarters.

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153)

.1 Reactor Scram on Simulated Water Level 2

a. Inspection Scope

The inspectors responded to the manual scram that occurred on November 15, 2007. The inspectors discussed the scram with operations, engineering, and licensee management personnel to gain an understanding of the event and assess follow-up actions. The inspectors reviewed operator actions taken in accordance with licensee procedures and reviewed unit and system indications to verify actions and system responses were as expected. The inspectors discussed the scram with the licensee's transient analysis procedure team and assessed the team's actions to gather, review, and assess information leading up to and following the scram. The inspectors later reviewed the final transient analysis report to assess the detail of review, adequacy of the apparent cause, and proposed corrective actions prior to unit restart.

b. Findings

Introduction: A Green self-revealed NCV of TS 5.4.1.a was identified for the failure to perform a maintenance activity in accordance with the documented instructions, which resulted in an unplanned scram from 9 percent power.

Description: During unit startup from RF12, operators were attempting to place the Division I reference leg backfill system in service on November 14. During the restoration activities, maintenance personnel were unable to achieve the desired backfill flow in accordance with Procedure 46.000.046, "Operation of the Reactor Reference Leg Backfill System." Maintenance staff determined that metering valve B21F520A was not functioning properly and generated WO 26198104 to replace the valve. Operations generated shift tagging record 2007-004712 to tag out the system so maintenance could replace B21F520A.

At 0312 on November 15, an operator performed step 1 of the safety tagging record which was to tag the backfill system isolation valve B2100F241A "closed." Because the Division I backfill system was not yet in operation, the valve was already closed so the operator attempted to verify the position. In performing this task, the operator briefly opened and then immediately closed the valve, contrary to the expectation to check valves expected to be closed in the closed direction only.

As soon as the operator manipulated this valve, he heard noises he believed to be associated with a reactor scram. Operators in the control room immediately noticed control rods starting to insert and that both reactor recirculation pumps tripped. Operators stabilized the plant and initiated CARD 07-27395 to investigate the cause of the scram.

The licensee determined the brief opening of B2100F241A caused a pressure perturbation on the instrument line which was connected to the high side of Division I level transmitters. The result was an increase in the differential pressure between the reference leg pressure and the low side of each transmitter which translated into an indicated reactor water level drop. Upon review of the computer logs, the pressure transient lasted for approximately 47 milliseconds which was just long enough to cause the Division I alternate rod insertion logic to actuate on a sensed reactor water level 2. However, it was not long enough for other logic systems to actuate, such as High Pressure Coolant Injection, RCIC, and the RPS, all of which have an actuation time of between approximately 100 and 250 milliseconds.

Because the pressure perturbation did not last long enough for RPS to initiate a reactor scram, the scram air header began to slowly depressurize on the alternate rod insertion actuation which caused control rods to start inserting. Based on the amount of time the false low reactor water level signal was in compared to the actual response time of the various instrument loops connected to the reference leg, the inspectors concluded the plant responded appropriately to the event.

<u>Analysis</u>: The inspectors determined the failure to properly operate B2100F241A only in the closed direction was a performance deficiency because it was contrary to licensee training and expectations. The inspectors determined the finding was more than minor in accordance with IMC 0612, Appendix B, Section 3, dated September 30, 2005. The inspectors compared this issue with the examples contained in Appendix E and determined example 4.b is related to this issue because the inappropriate valve operation led to a reactor scram. The inspectors determined this finding affected the Initiating Events Cornerstone and was of very low safety significance because, although a reactor scram occurred, it had no affect on the availability of mitigation equipment or functions. Immediate corrective actions included stabilizing the plant, reviewing the plant response to the scram to ensure all safety systems operated as designed, and re-emphasizing the importance and expectation that valves only be checked in the closed direction. The inspectors determined the finding was associated with a cross-cutting aspect in the area of Human Performance, Work Practices, because licensee personnel failed to comply with procedures when manipulating plant equipment (H.4(b)).

Enforcement: TS 5.4.1.a requires the licensee to establish, implement, and maintain written procedures as recommended in Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978. Regulatory Guide 1.33, Appendix A, Section 9, specifies that maintenance which can affect the performance of safety-related equipment be properly preplanned and performed in accordance with instructions appropriate to the circumstances. Contrary to this, on November 14, 2007, the licensee failed to properly perform a valve tagging activity in accordance with the documented instructions which had an actual effect on the performance of safety-related equipment. Because this finding was of very low safety significance, and because it was entered into the licensee's corrective action program as CARD 07-27395, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy:
NCV 05000341/2007006-09, Improper Valve Operation Caused Reactor Scram.

.2 Inadvertent RPS Undervoltage Trip

a. Inspection Scope

The inspectors responded to the RPS 'B' alternate breaker (EPA) trip that occurred on October 23, 2007. The inspectors discussed the breaker trip with operations, engineering, and licensee management personnel to gain an understanding of the event and assess follow-up actions. The inspectors reviewed operator actions taken in accordance with licensee procedures and reviewed unit and system indications to verify actions and system responses were as expected.

b. Findings

Introduction: A Green self-revealed NCV of 10 CFR 50.65(a)(4) was identified for the failure to appropriately assess the risk associated with removing RPS from its normal power source.

Description: On October 23, 2007, at approximately 0100, Division II RHR was being placed in service in the shutdown cooling mode. At the time, RPS 'B' was being powered from its alternate power supply because the RPS 'B' MG set was removed from service for maintenance. RPS 'A' was also removed from service for maintenance. The licensee started RHR pump 'D' which caused a voltage drop on the electrical bus powering RPS 'B' because the bus was heavily loaded at the time. Consequently, RPS tripped on the under voltage condition. The trip of the RPS breaker caused the following to occur:

- full reactor scram (all control rods were already inserted because of the outage);
- RHR shutdown cooling isolation;
- loss of RWCU;
- torus water management system isolation; and
- secondary containment isolation.

Operators entered Procedure 23.800.07, "Reactor Coolant Natural Circulation and Decay Heat Removal Methods," and verified the natural circulation mode was still capable of performing its decay heat removal function without RWCU in service. RWCU was restored using Abnormal Operating Procedure 20.707.01, "Loss of RWCU," and operators reset RPS 'B' alternate EPA breaker and re-energized RPS 'B'. Operators then restored the other affected systems.

The condition of starting RHR pump 'D' while the RPS was on alternate power is rare. However, the action of RPS 'B' alternate EPA breaker tripping when starting large motors in the plant is not rare. The licensee's corrective action program documents similar issues with this equipment six times between 1994 and 2001 and twice during RF12. Standard Operating Procedure 23.316, "RPS 120V AC and RPS MG Sets," contained a precaution in step 3.3 that stated, "…This is due to the susceptibility of loss of the RPS Alternate Transformer 'A' ('B') due to voltage transients on its power supply." This precaution was removed from revision 35 in 1997 because, "Engineering has shown that all EPA trips over the last nine years were the result of equipment failures and the EPA was performing as designed. This is no longer a commitment."

As an interim measure to preclude repetition, the licensee placed the precaution back in procedure 23.316 and was evaluating other corrective actions including a potential design change to the RPS alternate electrical power supply.

Analysis: The inspectors determined that the failure to perform an adequate risk assessment of the RPS maintenance was a performance deficiency because the licensee is expected to comply with 10 CFR 50.65(a)(4). The inspectors determined the finding was more than minor in accordance with IMC 0612, Appendix B, section 3, dated September 30, 2005. The inspectors compared this issue with the examples contained in Appendix E of IMC 0612 and determined that example 7.f is related to this issue. From this example, the inspectors concluded the issue was more than minor because had the risk assessment correctly recognized the potential for the loss of a shutdown key safety function, additional risk management actions to preclude such a loss would have been prescribed.

The inspectors determined that this finding affected the Mitigating Systems Cornerstone. The inspectors assessed the finding using IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," dated February 28, 2005. The inspectors referred to Attachment 1, Checklist 7 because reactor coolant system level was greater than 23 feet above the flange. The inspectors determined this finding affected the equipment portion of the core heat removal guidelines but answered, "No" to all three screening questions for findings requiring a phase 2 or phase 3 analysis. Specifically, this finding was only related to the loss of decay heat removal but did not degrade the licensee's ability to recover decay heat removal. Using Appendix G, Table 1, the inspectors concluded that the finding was not related to a loss of thermal margin because the temperature margin to boil was approximately 0.01°F. Therefore, using Appendix G, Figure 1, the inspectors concluded this finding did not require a quantitative assessment and therefore screened as Green.

Enforcement: 10 CFR 50.65(a)(4) requires, in part, that the licensee assess and manage the increase in risk that may result from a maintenance activity prior to performing the activity. Contrary to the above, on October 23, 2007, the licensee started the 'D' RHR pump with the RPS 'B' on the alternate power supply during maintenance which caused the loss of a shutdown key safety function. The failure to adequately assess risk of starting the RHR pump while performing maintenance is a violation of 10 CFR 50.65(a)(4) in accordance with paragraph 7.11.1.d.1(b)(4) of the NRC Enforcement Manual. Because this finding was of very low safety significance and because it was entered into the licensee's corrective action program as CARD 07-26537, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000341/2007006-10, Inadvertent RPS Undervoltage Trip.

.3 Safety Relief Valve Tailpipe Hole

a. Inspection Scope

On Thursday, October 11, 2007, with Fermi 2 in Mode 5 for a refueling outage, licensee personnel identified what appeared to be a through-wall hole and other suspicious indentations of varying depths on several safety relief valve tailpipes. The hole and indentations appeared to have been caused by drilling activities. Based on initial available information, the licensee determined that the potential for tampering existed. In accordance with the Fermi 2 Emergency Plan, the licensee declared a Notice of Unusual Event in accordance with Emergency Action Level HU4. Region III dispatched a Special Inspection Team to review the event. The findings and conclusions of the team are in Inspection Report 05000341/2007010.

.4 Division I Emergency Diesel Generator Load Sequencer Failure – October 4

a. Inspection Scope

The inspectors reviewed the events and circumstances surrounding the loss of Division I EDG load sequencer which rendered both Division I EDGs inoperable at the same time that both Division II EDGs were inoperable due to maintenance. The inspectors reviewed the subsequent operator actions to ensure appropriate licensee procedures and TSs were followed. The inspectors discussed the licensee's troubleshooting results with operations, management, and members on the troubleshooting team to fully understand the cause of the failure and proposed corrective actions. The inspectors also reviewed the reportability requirements specified in NUREG-1022, Event Reporting Guidelines, to determine if this issue constituted a reportable event under 10 CFR 50.72 or 10 CFR 50.73.

The licensee determined that a 24-volt power supply unit in the load sequencer failed which caused the failure of the entire load sequencer. With the load sequencer unable to perform its function, required post-accident loads would not automatically sequence back on at the prescribed times. However, since the plant was operating in Mode 5 with the plant depressurized, there was no equipment that would have been required to automatically sequence back on after any postulated accident. For example, if a loss-of-offsite power would have occurred, the running decay heat removal pumps would have tripped; however, licensee procedures required the pumps to be manually restarted after the discharge piping had been verified to be filled and vented. All other equipment normally required to automatically sequence back on after an accident was not required by the TSs to be operable in Mode 5. Therefore, the inspectors agreed with the licensee's assessment that this issue did not constitute a reportable event.

b. Findings

No findings of significance were identified.

.5 <u>(Closed) LER 05000341/2007001, Excessive Feedwater Check Valve Leakage at Containment Penetration</u>

On October 7, 2007, B2100F076B, Feedwater Line ‘B’ Outboard Check Valve, and B2100F010B, Feedwater Line ‘B’ Inboard Check Valve, failed as-found LLRT leak rate testing during RF12 due to excessive seat leakage. The minimum penetration leakage resulting from these failures, 297.3 standard cubic feet per hour, exceeded the maximum TS allowable containment leakage of 296.3 standard cubic feet per hour. The licensee entered the failures into their corrective action program as CARD 07-25836 and performed a root-cause evaluation. The licensee determined the current soft seats utilized in all four feedwater check valves were unreliable beyond one operating cycle. Consequently, the licensee installed new soft seats for all four feedwater check valves and revised the requirement to change the soft seats from every other refueling outage to every refueling outage. By the end of this inspection period, the licensee was still reviewing the root and contributing causes to determine an appropriate action plan.

As described in Section 4OA2.4 of inspection report 05000341/2007004, the inspectors determined the licensee’s root-cause evaluation of the LLRT failures of the ‘A’ feedwater check valves in RF11 and the corresponding corrective actions implemented during that outage were inadequate. An NCV against 10 CFR 50, Appendix B, Criterion XVI, was documented. During this current inspection period, the inspectors determined the failure of check valves B2100F076B and B2100F010B and subsequent failure to maintain total containment leakage within TS requirements prior to RF12, directly resulted from the failure to implement adequate corrective actions following the failures in RF11. The performance deficiency that led to the failure to maintain total containment leakage within TS requirements in RF12 was already addressed by NCV 05000341/2007004-05. Further corrective actions for this additional example are expected to be taken in conjunction with corrective actions for the previous NCV. No additional findings were discovered while reviewing this LER. This LER is closed.

4OA5 Other Activities

.1 <u>Inspection of Licensee Strike Contingency Plans (92709)</u>

a. <u>Inspection Scope</u>

The licensee entered this inspection period without a labor agreement with the local union representing maintenance, chemistry, warehouse, and fire protection staff as well as non-licensed operators. The licensee and union were operating under a signed agreement that contract negotiations would occur without possibility of a work stoppage unless either party provided a 45-day written notice of intent to cancel the agreement. On October 25, the union provided such a 45-day notice which would have placed the earliest possible work stoppage at December 9, 2007. On November 26, the inspectors began evaluating the adequacy of the licensee’s strike contingency plan to determine if:

- the required minimum number of qualified personnel were available for the proper operation and safety of the facility;
- reactor operation and facility security would be maintained as required; and,
- the plan complied with the requirements in TS, Emergency Plan, and the CFR.

Specifically, the inspectors reviewed the plan, TS, Emergency Plan staffing requirements, procedures, and other documents, and interviewed personnel to:

- determine the adequacy of the licensee's strike contingency plans and that those plans had been reviewed by the appropriate level of licensee management;
- determine if the licensee could meet the requirements for minimum onsite shift staffing of the facility;
- determine if the licensee could meet regulatory requirements in the areas of plant management, operations, maintenance, security, chemistry, RP, surveillance and calibrations, and administrative controls;
- verify the licensee properly trained licensed personnel who would be performing functions to which they are not normally assigned;
- verify through observation and discussion with at least one person from plant management, operations, maintenance, RP, and chemistry to ensure they understood their function under the modified staffing plan;
- verify that support from local agencies was adequate to ensure unimpeded access of personnel to the plant, unencumbered delivery of support goods to the site and unencumbered offsite shipment of radioactive materials, mitigation of any possible threat to the site including abusive or violent strikers, unimpeded access of medical care and ambulance services to treat injured or contaminated persons, and unimpeded access of the local fire department to supplement the site fire fighting unit;
- confirm that site staffing would be sufficient and qualified to implement the site emergency plan; and
- verify that the emergency communication equipment and the Emergency Notification System, where applicable, are available and operable.

The licensee and union reached a tentative contract agreement that was ratified by the union on December 7; therefore, no work stoppage occurred prior to the new contract approval. Concerns that the inspectors identified with the licensee's strike contingency plans were adequately addressed prior to the contract ratification.

b. Findings

No findings of significance were identified.

.2 (Closed) URI 05000341/2007003-05: Inrush Current of Spring Charging Motors Not Considered.

During the Component Design Bases Inspection the inspectors identified a URI pertaining to calculation DC-0213, Revision Q, "Sizing of 130/260V Batteries." Specifically, the calculation did not consider the inrush current of the spring charging motors, associated with closing mechanism of 4160V and 480V switchgear circuit breakers, in determining the battery's 1-minute rating. The licensee recognized this condition in 2003 and issued a CARD to incorporate the inrush current of spring charging motors into the calculation and re-evaluate the battery's 1-minute rating.

During a review of battery sizing calculation DC-0213, the inspectors identified that the calculation used average current values instead of inrush current values of spring charging motors associated with closing mechanism of 4160V and 480V switchgear circuit breakers. Per Institute of Electrical and Electronic Engineers (IEEE)

Standard 485-1997, “IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications,” momentary loads such as the switchgear operations and inrush currents should be used when determining battery’s 1-minute rating. Momentary loads could occur one or more times during the duty cycle, but would be of short duration, not exceeding 1-minute at any occurrence. Although momentary loads may exist for only a fraction of a second, it is common industry practice to consider that each load would last for a full minute because the battery voltage drop after several seconds often determined the battery’s 1-minute rating. Sizing for a load lasting only a fraction of a second, based on the battery’s 1-minute performance rating, would result in a conservatively sized battery. When several momentary loads occur within the same 1-minute period and a discrete sequence cannot be established, the load for the 1-minute should be assumed to be the sum of all momentary loads occurring within that minute. During the minute, depending on how many momentary loads occur, the inrush current pulls the battery voltage down and, therefore, it would be necessary to ensure the battery was adequately sized to provide the required voltage to the loads. Because of the failure to consider the inrush current in sizing the batteries, the inspectors were concerned that the batteries might not have been adequately sized to provide the required voltage to the loads.

In light of the above described condition, the licensee took corrective actions and revised the battery sizing calculation using starting current of the spring charging motors and also performed a past operability determination for the safety-related batteries.

Based on the inspectors’ review of the revised calculation DC-0213, Revision T, the inspectors determined that even though the margin decreased slightly, there was still adequate margin left for the operability of both the Division I and Division II safety-related batteries. As such, this unresolved item is closed.

This inspection activity does not represent an inspection sample for this report.

4OA6 Management Meetings

.1 Exit Meeting Summary

On January 17, 2008, the inspectors presented the inspection results to Mr. J. Davis and other members of the licensee staff. The licensee acknowledged the issues presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. All proprietary information obtained from the licensee during this inspection was returned to the licensee.

.2 Interim Exit Meetings

Interim exit meetings were conducted for:

- Inspection Procedure 71111.08, “In-service Inspection Activities” with Mr. M.Caragher and other members of the licensee's staff on October 12, 2007;
- Occupational Radiation Safety Program for access control to radiologically significant areas with Mr. J. Plona on October 9, 2007, and with Mr. T. Brown on November 1, 2007;
- Public Radiation Safety Program for Radioactive Material Processing and Transportation and Performance Indicator Verification with Mr. Kevin Hlavaty on December 14, 2007;

- Emergency Preparedness Inspection with Mr. G. Garber on December 19, 2007;
- Biennial Operator Requalification Inspection, except for annual requalification test results with Mr. J. Plona, Site Vice President and other members of the licensee's staff on December 21, 2007 ;
- Biennial Operator Requalification Inspection results associated with the biennial written examination and annual operating test results with Mr. D. Coseo on January 2, 2008; and on January 16, 2008, inspectors presented additional information, to Mr. G. Baustian, related to the issue associated with biennial written exam failure rate.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

J. Davis, Chief Nuclear Officer
J. Plona, Site Vice President
K. Hlavaty, Director Nuclear Production
C. Walker, Director Organizational Effectiveness
M. Caragher, Director Nuclear Engineering
G. Baustian, Manager Nuclear Training
T. Brown, Manager Radiation Protection
T. Dong, Manager Performance Engineering
G. Garber, Radiological Emergency Response Specialist
R. Gaston, Manager Licensing
K. Howard, Manager Plant Support Engineering
J. Janssen, Manager Maintenance
E. Kokosky, Manager Performance Improvement
J. Moyers, Manager Nuclear Quality Assurance
D. Noetzel, Manager Engineering First Team
K. Scott, Manager Nuclear Operations
K. Snyder, Manager System Engineering Manager
T. Stack, Manager Nuclear Security

Nuclear Regulatory Commission

C. Lipa, Chief, Reactor Projects Branch 4

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000341/2007006-02	URI	Undocumented Technical Basis for Change to EOP (Section 1R11.4)
05000341/2007006-07	URI	Non-Qualified Ty-Wraps Inside Primary Containment (Section 1R20.2)

Opened and Closed

05000341/2007006-01	NCV	Failure to Expand Scope of Pipe Support Hanger Examinations (Section 1R08.1)
05000341/2007006-03	FIN	High Failure Rate on Licensed Operator Requalification Examination (Section 1R11.10)
05000341/2007006-04	NCV	Inadequate Work Planning for Shield Block Wall Removal (Section 1R13.1)

05000341/2007006-05	NCV	Failure to Maintain Configuration Control of the RCIC Pump (Section 1R19.1.b(1))
05000341/2007006-06	FIN	Transient During Testing of #5 Low Pressure Stop Valve (Section 1R19.1.b(2))
05000341/2007006-08	NCV	A HRA was entered without adequate awareness of radiological conditions and without an RWP that allowed HRA entry. (Section 2OS1.3)
05000341/2007006-09	NCV	Improper Valve Operation Caused Reactor Scram (Section 4OA3.1)
05000341/2007006-10	NCV	Inadvertent RPS Undervoltage Trip (Section 4OA3.2)

Closed

05000341/2007-001	LER	Excessive Feedwater Check Valve Leakage at Containment Penetration (Section 4OA3.5)
05000341/2007003-05	URI	Inrush Current of Spring Charging Motors not Considered (Section 4OA5.2)

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspector reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

<u>1R01 Adverse Weather Protection</u>

Miscellaneous:
- Cold Weather Preparation Checklist

<u>1R04 Equipment Alignment</u>

Drawings:
- 6M721-5706-1, Revision AA: Residual Heat Removal Division II, Functional Operating Sketch
- 6M721-5706-2, Revision W: Residual Heat Removal Division 1 Functional Operating Sketch
- 6M721-5734, Revision AW: Emergency Diesel Generator System Functional Operating Sketch

CARDs:
- CARD 06-23665: Packing Leak on E1150-F009; 04/29/2006
- CARD 06-26821: Evaluate As-Found Inspection Results for RHR A Motor; 10/23/2006
- CARD 06-27618: ECCS Pumps Are Not Insulated as Assumed in the Plant Design Basis; 11/20/2006
- CARD 07-22779: Component Description Discrepancies; 07/06/2007
- CARD 07-22323: Send Motor Back to Westinghouse for Vibration Testing; 05/15/2007
- CARD 07-22323: Send Motor Back to Westinghouse for Vibration Testing; 05/15/2007

Procedures:
- Procedure 23.205, Revision 98: Residual Heat Removal System; 03/09/2007

Work Orders:
- WO25744724: Add Interposing Relays for E1150F017B; 10/17/2007

<u>1R05 Fire Protection</u>

CARDs:
- CARD 07-26334: NRC Concern – Fire Watch Actions; 10/18/2007 (NRC-Identified)
- CARD 07-26521: NRC Concern Main Unit Transformer 2A Deluge Piping; 10/22/2007

Procedures:
- Procedure FP-RB-1-7b, Revision 4: Reactor Building South Control Rod Drive (CRD) and Railroad Bay Area, Zone 7; 06/12/2006
- Procedure FP-RB-1-7a, Revision 4: Reactor Building North Control Rod Drive (CRD) Area, Zone 7; 05/02/2006
- Procedure FP-RB-4-17b, Revision 3: Reactor Building Recirculation System Motor Generator Area, Zone 17; 01/25/2000

Miscellaneous:
- Selected Fire Extinguisher Inspection Tags
- UFSAR Section 9A.4, Fire Protection Program

1R06 Flooding

CARDs:
- CARD 05-25383: SEN Internal Flood Design Deficiencies; 09/22/2005

Procedures:
- Procedure 20.000.03, Revision 9: Turbine Building Flooding; 07/22/2005
- Procedure 28.507.05, Revision 15; Inspection of Penetration Fire Stops; 08/30/2006

1R07 Heat Sink Performance

Work Orders:
- WO #W840070100: Perform 18 Month Inspection on EDG-12; 12/17/2007

1R11 Licensed Operator Requalification Program

CARDs:
- CARD 07-28195, EOP Difference Document Does Detail Terminate and Prevent in 29.ESP.01; 01/20/2007 (NRC-Identified)
- CARD 08-20103, Evaluate High Failure Rate on 2007 Requalification Written Examination; 01/08/2008 (NRC-Identified)
- Condition Reports Related to Licensed Operator Requalification Training; Various; dated 2006 – 2007

Reports:
- Evaluation Summary Report (Simulator) - Student Feedback; Various; 2006 - 2007
- Evaluation Summary Report (Simulator) - Management Observation of Training; Revision 1/16/04; Various; 2006 - 2007
- Evaluation Summary Report - Simulator Instructor Observation; Revision 7/1/05; Various; 2006 2007
- Evaluation Summary Report (Classroom) - Student Feedback; Revision 5/15/06; Various; 2006 - 2007
- Evaluation Summary Report (Classroom) - Management Observation of Training; Revision 1/16/04; Various; 2006 - 2007
- Evaluation Summary Report-Classroom Instructor Observation; Revision 7/1/05; Various; 2006 - 2007
- Licensed Operator Requalification Program Results; 2006-2007
- Licensed Operator Curriculum Review Committee Meeting Minutes; Various; 2006 - 2007
- Licensed Operator Requalification Annual Operating Examinations - Various; 2007
- Licensed Operator Requalification Biennial Written Examinations - Various; 2007
- Licensed Operator Requalification 24-Month Training Plan; 2006 - 2007
- Licensed Operator Requalification Training Cycle Attendance Records – Various; 2006 - 2007
- ODE-8; Attachment 3; Shift (1-5) Active License Required Hours (1st - 4th) Quarters 2006 and (1st - 3rd) Quarters 2007

- ODE-8; Attachment 3; Off Shift Active SRO (RO) License Required Hours (1st - 4th) Quarters 2004 and (1st - 3rd) Quarters 2005
- Simulator Differences; Various; 2007

Procedures:
- 29.100.01, Sheet 1A, RPV Control – ATWS; Revision 10
- 29.ESP.01, Supplemental Information, Enclosure A; Revision 14
- CP-OP-232, Annual Requalification Examination; Revision 11
- EOP Support Documentation; Revision 21
- Nuclear Training Work Instruction 5.12, Conduct of Simulator Assessments and Evaluations; Revision 8
- Nuclear Training Work Instruction 5.15, Remediation and Re-evaluation; Revision 6
- Operations Department Expectation (ODE) 8, Administrative Guidelines and Desk Instructions; Revision 5
- Simulator Maintenance Procedure SM-8; Attachment 2; Malfunction Event Performance Test Data Sheet; Event Number 014; Loss of Both Recirculation Pumps; Revision 8
- Simulator Maintenance Procedure SM-8; Attachment 2; Malfunction Event Performance Test Data Sheet; Event Number 0008; Reactor Manual Scram; Revision 10
- Simulator Maintenance Procedure SM-7; Operations Procedures Performance Test; Startup from Cold Shutdown to Hot Standby; Revision 13

Other:
- BWR Owners Group Emergency Procedure and Severe Accident Guidelines, Appendix B, Technical Basis; Revision 2
- Fermi 2 Evaluation Scenario: SS-OP-904-1051
- Fermi 2 Evaluation Scenario: SS-OP-904-0008

1R12 Maintenance Effectiveness

CARDs:
- CARD 07-25700: Division I EDG Sequencer Trouble Alarm Received; dated 10/04/2007

Miscellaneous:
- Maintenance Rule Functional Failure Evaluation 070717-01-13: CARD 07-23962; dated 07/31/2007
- Selected Operator Logs From January 1, 2005, through October 1, 2007
- Selected Emergency Diesel Generator Maintenance Rule Out-of-Service Evaluations

1R13 Maintenance Risk Assessments and Emergent Work Control

CARDs:
- CARD 97-12161: UFSAR Descriptions for TBHVAC and RBHVAC Do Not Support On-Line System Outages
- CARD 00-14755: Elevated TB Temps, Contingency Action Recommendation; 08/07/2000
- CARD 06-27610: Remove TB-1 Steam Tunnel Shield Block Walls to Support Condenser Pump Work in RF 12; dated 11/30/2006
- CARD 07-25241: Early Recovery from TBHVAC Outage Due to TB3 Conditions; dated 09/18/2007
- CARD 07-25277: Investigate Affect of Opening TB 1st Floor Steam Tunnel on TB 2 Steam Tunnel Temperatures; dated 09/20/2007
- Div 1 EDG Sequencer Trouble Alarm Received; dated 10/04/2007

Drawings:
- 6SD721-2530-10 AF: 260/130V ESS Dual Battery; 2 PA Distribution – Division I
- 6I721-2714-22 F: EDG Automatic Digital Load Sequencing System H11P898A

Procedures:
- Procedure 23.800.07, Revision 9: Reactor Coolant Natural Circulation and Decay Heat Removal
- Procedure 23.412, Revision 44; Turbine Building Heating, Ventilation, and Air Conditioning System

Miscellaneous
- Apparent Cause Evaluation Template and Self-Check List for CARD 07-25700; dated 10/12/2007
- CDF Risk Profile for 11/18 to 11/26; dated Week of 11/19/2007
- Class 1E Fuse List;
- DTE Nuclear Generation Memorandum TMSA-07-0004: RF12-M43: Adequacy of Decay Heat Removal Methods; 02/21/2007
- DTE Nuclear Generation Memorandum TMSA-07-0028: Time-to-Boil Calculations for RF-12
- Operator Log from 10/04/2007 to 10/05/2007
- Root Cause Determination for CARD 07-25277: Investigate Affect of Opening Turbine First Floor Steam Tunnel on the Turbine Building Steam Tunnel Temperatures; dated 10/29/2007
- Shutdown Cooling Outage Contingency Plans
- Scheduler's Risk Evaluation for Fermi 2, Schedule as of 11/21/2007 16:28

1R15 Operability Evaluations

CARD:
- CARD 07-26464: E5150F008 Cable Inspection
- CARD 07-26974: Ty-Wraps in Drywell

1R17 Permanent Plant Modifications

Miscellaneous
- Engineering Design Package 33703: EECW Pump and Motor Replacement; August 24, 2007
- Engineering Change Request 33703-1: EECW Pump Replacement Modification; September 12, 2007
- Engineering Change Request 33703-2: EECW Pump Replacement Modification; October 11, 2007
- Engineering Change Request 33703-3: Document Lug Modification and Raychem Repair for P4400C001 A and B; October 11, 2007

1R19 Post Maintenance Testing

CARDs:
- CARD 07-25836: LLRT Failure of B2100F076B Exceeds La; 10/07/2007
- CARD 07-26848: Head Correction Not Applied to EECW Differential Pressure Switch ICSS; 10/31/2007
- CARD 07-27117: RF12 PCILRT Verification Test Phase Criteria Not Met: 11/08/2007
- CARD 07-27080: Unexpected Increase in RBCCW Head Tank Level; 11/07/2007

- CARD 07-27462: Failure to Meet Acceptance Criteria During RCIC System Pump and Valve Operability Test; dated 11/17/2007
- CARD 07-27464: Procedure 24.206.01 Surveillance Needs Revision; dated 11/17/2007
- CARD 07-27478: Human Performance Error Results in RPV Water Level Transient and Turbine Bypass Valve Stroke; 11/17/2007
- CARD 07-27469: High Severity Wear Index on RCIC Pump Bearings; dated 11/17/2007
- CARD 07-27579: RCIC Pump Bearing Oil Sample High Severe Wear Index; dated 11/22/2007
- CARD 07-25974: RCIC Thrust Bearing was Determined to be Different Than What Was Originally Specified in the Vendor Manual; dated 10/10/2007

Procedures:
- Procedure 24.137.01, Revision 35: Main Steam Line Isolation Channel Functional Test
- Procedure 24.137.03, Revision 38: Main Steam Line Valve Operability Test
- Procedure 24.207.08, Revision 68: Division 1 EECW Pump and Valve Operability Test
- Procedure 24.207.09, Revision 29: Division 2 EECW Pump and Valve Operability Test
- Procedure 35.206.003, Revision 28: RCIC Pump Rotating Assembly Removal and Installation
- Procedure 35.809-002, Revision 29: Horizontal Rotating Equipment Alignment
- Procedure 43.000.05, Revision 31: Visual Examination of Piping and Components (VT-2)
- Procedure 43.401.00, Revision 28: Integrated Leak Rate Test – Type A - General
- Procedure 47.000.03, Revision 24: IST Pump Reference Value Testing Instruction

Work Orders:
- WO # 000Z033471: Cutout and Replace Valve G3300F120; 10/08/2007
- WO # 000Z052900: RHR Pump A Motor Vibration Replace; 10/18/2007
- WO # 000Z053614: Degraded Grease in MOV. Refurbish Actuator in RF12; 09/27/2007
- WO # 25134233: Replace EECW Div II Pump per EDP-33703; 08/02/2007
- WO # 26221144 Impact Statement; dated 11/26/2007
- WO # E520961116: Perform 10 Year Internal Inspection of RCIC Pump; 03/31/2007
- WO # 25129754, Replace EECW Division I Pump Per EDP-33703
- WO # 1138030328, Perform 43.401.100 Type A PCIRLT
- WO # 0984070929, Perform 24.137.21 Reactor Pressure Vessel System Leakage Test

Miscellaneous:
- DTE Purchase Order 055947: Impeller; dated 02/03/1982
- DTE Purchase Order NR-319003: dated 10/10/1996
- EDP 35464, RCIC Pump Impellers Reinstallation; dated 11/20/2007
- Equivalent Replacement Evaluation 33926: RCIC Pump Rotating Assembly Replacement; dated 04/26/2007
- IST Response to CARD 07-26617: NRC Concern – Procedure 24.207.09, Ultrasonic Flow Measurement Range (NRC-Identified)
- Surveillance Performance 0383090401: Perform 43.401.303 LLRT for X-9A; dated 11/19/2007
- Surveillance Performance 0268071120: Perform 24.206.01 RCIC System Pump Operability and Valve Test @ 1000 PSIG; dated 11/17/2007
- Surveillance Performance 0384070928: Perform 43.401.304 LLRT for X-9B; dated 10/18/2007
- Surveillance Performance 0866070320: Perform 64.120.040 Containment Area High Range Radiation Monitor Division 1 Electrical Calibration; 08/30/2007

- Surveillance Performance 0867070320: Perform 64.120.041 Containment Area High Range Radiation Monitor Division 2 Electrical Calibration; 08/17/2007
- Surveillance Performance 3272011028: Perform 24.204.01 Division 1 LPCI and Torus Cooling/Spray Pump and Valve Operability Test; 11/01/2007
- Test Review and Approval Request SOE No.: 07-02, Revision 1: Perform In-Service Testing of the New Division 2 EECW Pump and Motor for Pump Curve Verification for EDP 33703; 10/26/2007
- Technical Evaluation TE-B21-07-062: Design Basis for Using Soft Seat Material for Cycle 13; dated 11/09/2007
- Work Request # D783090100, Revision 1: Moved Leak Check of Air Lines form Job Instructions to PMT; 10/18/2007
- Work Request # E520930223: Perform 10-Year Internal Inspection of RCIC Sulzer-Bingham Pump; 07/20/2006

1R20 Outage Activities

CARDs:
- CARD 07-25588: Division 2 Core Spray Injection Check Valve Failed LLRT; 10/02/2007
- CARD 07-25500: Leakage Discovered in Subpile Room; 09/29/2007
- CARD 07-25613: Inadvertent Manual Closure of Division 2 EESW Pump Breaker; 10/02/2007
- CARD 07-26214: Drywell Cooler #14 Inlet Side of “A” Coil Gasket Leaking; 10/16/2007
- CARD 07-26537: RPS B Alternate EPA Breaker Tripped When RHR Pump D Was Started; 10/23/2007
- CARD 07-26742: G1154F600 Failed LLRT per 43.401.314; 10/28/2007
- CARD 07-26784: Ineffective Corrective Actions for CARD 06-21751; 10/29/2007
- CARD 07-26785 Initial Response: NRC Identified Issues in Torus; 11/01/2007
- CARD 07-26974: Ty-Wraps in Drywell; 11/03/2007
- CARD 07-26887: Degraded Protective Coating in Drywell; 11/02/2007
- CARD 07-26904: Abnormal Elongation Readings Following Tensioning; 11/01/2007
- CARD 07-26908: Need Specific Coating Evaluation Prior to Drywell Closeout; 11/01/2007
- CARD 07-26926: Debris Being Flushed into Drywell/Reactor Building Floor Drain Sumps; 11/02/2007
- CARD 07-27197: Model of ECCS Suction Strainer Debris Loading Due to Shield Blankets Outside Range of Method Applicability; 11/09/2007
- CARD 07-27455: Parts of Insulation Missing on Valve; dated 11/16/2007
- CARD 07-27456: Insulation Walked On and Is Not Installed Against Piping. Banding Very Loose; 11/16/2007
- CARD 07-27457: Materials Found During Leak Walkdown in RB-1 Steam Tunnel; dated 11/16/2007
- CARD 07-27459: Safety Near Miss. 3-Foot Lead Blanket Burned (Left on Valve G3300F121) During Start-up; dated 11/16/2007

Procedures:
- Procedure 22.000.02, Revision 68: Plant Startup to 25% Power

Miscellaneous:
- Design Specification 3071-389: Emergency Core Cooling System (ECCS) Suction Strainers; 07/31/1998
- Deviation Event Report 93-0255: Potential Plugging and Collapsing of RHR Suction Strainers; June 8, 1993

- Engineering Functional Analysis E11-07-005: Non-Qualified, Tefzel Ty-Wraps Inside Primary Containment; 11/30/2007
- Executive Summary for the Resolution of the ECCS Strainer Issues (CARD 07-27197)
- IPCS Data for Level, Pressure and Power – Initial Condition; 11/17/2007
- IPCS Date for Level, Pressure, and Power – Transient Min-to-Peak; 11/17/2007
- Letter: NRC 94-0035, Response to NRC Bulletin 93-02, Supplement 1; April 19, 1994
- Letter: NRC 94-0095, Confirmation of Completion of the Requested Actions for NRC Bulletin 93-02, Supplement 1; October 17, 1994
- Letter: NRC 98-0145, 120 Day Response to Generic Letter No. 98-04; November 11, 1998
- RF12 Drywell Entry Action List
- Radiological Surveys, Drywell
- Radiation Work Permit 07-1054, Revision 1: Drywell Initial Entry for Outage FR 12 and Leak Walkdowns; 09/28/2007
- Risk Management Plan: RF-12 Fuel Loading with SRM A and SRM B Inoperative; 10/17/2007
- Technical Evaluation TE-R34-07-070: Ty-Wraps in Drywell

1R22 Surveillance Testing

CARDs:
- CARD 07-25816: B2100-F010B Failed LLRT; 10/06/2007
- CARD 07-25836: LLRT Failure of B2100F076B Exceeds La
- CARD 07-27028: Division 2 130 VDC Battery Charger 2B-1 Would Not Reset; 11/05/2007

Procedures:
- Procedure 24.307.02, Revision 42: Emergency Diesel Generator 12 – Loss of Offsite Power and ECCS Start with Loss of Offsite Power Test
- Procedure 24.307.03, Step 5.2, Revision 40: Emergency Diesel Generator 13 – ECCS Start with Loss of Offsite Power Test

Miscellaneous
- Surveillance Performance 0279071009: Perform 24.207.09 Section 5.1, Division II EECW Pump and Valve Operability Test; 10/09/2007
- Surveillance Performance 3272011028: Perform 24.204.01 Division 1 LPCI and Torus Cooling/Spray Pump and Valve Operability Test; 11/01/2007
- Surveillance Performance 3589071027: Perform 43.401.516 RHR Pressure ISO Valve Leakage (Test 2:E1100F050B); 10/02/2007
- Surveillance Performance RE37071023: Perform SFP Loading Evaluation; 10/19/2007
- Surveillance Performance 0383070920: Perform 43.401.303 LLRT for X-9A (Test-1:B2100F010A; 10/07/2007
- Surveillance Performance 9004071007 (25817S17): Perform 43.401-304 LLRT for X-9B (Test-1:B2100F010B)
- Master Core Loading Pattern; 10/17/2007

1R23 Temporary Plant Modifications

CARDs:
- CARD 07-27051: Power source for B.5.b Radios; 11/06/2007

Work Orders:
- WO 26228344: 04-Steam Leak on N30-F006, furmanite per TM 07-0026, dated 12/12/2007

Miscellaneous:
- MMA21, Revision 2: Temporary Power, Extension Cords, and Free Air Signal Cables
- Risk Management Plan: Furmanite Injection of Valve N30F006; dated 12/12/2007
- Temporary Modification 07-0021, Revision 0: Provide Temporary Power to R1700S016A during RF-12 72B Bus Inspection; dated 10/14/2007
- Temporary Modification 07-0026, Rev A; On-line Leak Repair of N30F006, MS to RHTRS Warm-Up PCV.

1EP4 Emergency Action Level and Emergency Plan Changes

- Fermi 2 Power Plant Emergency Plan, Revisions 31, 32, and 33

2OS2 ALARA Planning and Controls

CARDs:
- CARD 07-25796; High Radiation Area Entry on Incorrect Task; dated October 6, 2007
- CARD 07-26343; Stay Time Tracking Field Issues; dated October 18, 2007 (NRC Identified)
- CARD 07-26387; Improvement Item for Source Term Reduction; dated October 19, 2007 (NRC Identified)
- CARD 07-26388; Improvement Item for Dose Reduction; dated October 19, 2007 (NRC Identified)
- CARD 07-26358; CARD Significance Level Determination for Procedure Compliance Issues; dated October 18, 2007 (NRC Identified)
- CARD 07/26376; Respiratory Evaluation Worksheet Inconsistencies; dated October 19, 2007 (NRC Identified)
- CARD 07-26378; Stay Time Tracking Errors; dated October 19, 2007 (NRC Identified)
- RF-10 ALARA Assessment; Radiological Engineering; dated April 2005

Miscellaneous:
- RF-11 Post Outage ALARA Assessment; date not provided
- PTP 63.000.200; ALARA Reviews; Revision 22; dated September 18, 2007
- Radiation Protection Conduct Manual; MRP05; ALARA/RWPS; Revision 7; dated July 19, 2007
- Radiation Protection Conduct Manual; MRP04; Fetal Protection Program; Revision 4; dated July 24, 2007
- Fermi 2 ALARA Committee; Dose Reduction Plan; date not provided
- Radiation Work Permit 07-1251; Perform Refuel Activities on RB-5; Revision 0
- Quality Assurance Conduct Manual; MQA11; Condition Assessment Resolution Document; Revision 22; dated September 28, 2007
- PTP 63.000.100; Radiation Work Permits; Revision 26; dated September 21, 2007
- PTP 67.000.101; Performing Surveys and Monitoring Work; Revision 25; dated September 20, 2007
- Radiation Work Permit and associated documentation; RWP 07-1052; S. Reactor Water Cleanup Pump Repair; dated February 15, 2007
- Radiation Work Permit 07-1029; Replace South RWCU Pump Motor; Revision 7

2PS2 Radioactive Material Processing and Transportation

CARDs:
- CARD 05-21433; Shipping Survey Not Performed in Accordance with 67.000.103 for Shipment 05-006; dated March 3, 3005

- CARD 05-23253; Unused Liquid Radwaste Subsystems; dated May 25, 2007
- CARD 05-26147; Procedure MMM10 not Followed and Radioactive Material Package Opened before Survey; dated November 2, 2005
- CARD 06-23359; Hazmat Function Specific Training for New Hires; dated May 15, 2006
- CARD 06-27277; Lead Shielding Discovered in GE Shipping Containers Contrary to Type A Packaging Certificates; dated November 10, 2006
- CARD 07-23041; Vendor Supplied Type B Cask Arrived with Bad Gasket; dated 05/31/07
- CARD 07-26096; MRP04 Contains Ambiguous Requirements for Placing RAM Tags on Containers that Contain Radioactive Materials
- CARD 07-28019; Procedure Enhancements for 65.704.001 Setup and Operating
- CARD 07-28021; Unknown Epoxy Material on Waste Sample Tank B; dated 12/13/07; (NRC Identified)
- CARD 07-28020; Radwaste Equipment Status; dated December 13, 2007 (NRC Identified)
- CARD 07-28013; Warehouse A Loading Dock Rollup Door Left Open and Unattended Contrary to CARD 04-21072; dated December 13, 2007 (NRC Identified)

Procedure:
- 67.000.103; Survey of Outgoing Radioactive Material Shipments; Revision 19

Miscellaneous:
- Shipment Number 05-023; Dewatered Resins, Liner LH 03-005; dated April 19, 2005
- Shipment Number 05-039; Dewatered Resins, Liner LH 04-002; dated June 23, 2005
- Shipment Number 06-006; Contaminated Laundry; dated January 30, 2006
- Shipment Number 06-101; GE Fuel Inspection Equipment; dated November 21, 2006
- Shipment Number EF2-07-034; Dewatered Resins; dated June 1, 2007
- Shipment Number EF2-07-123; Contaminated Laundry; dated December 13, 2007
- RDS-1000; dated December 13, 2007 (NRC Identified)

<u>4OA1 Performance Indicator Verification</u>

Miscellaneous:
- Maintenance Rule Functional Failure Evaluation 070717-01-13: CARD 07-23962; dated 07/31/2007
- MSPI, Cooling Water Systems 4Q/05 through 2Q2007 Derivation Report
- MSPI, Emergency AC Power System 4Q/05 through 2Q2007 Derivation Report
- Operator Log dated 01/03/2007 through 09/28/2007
- Reactor Coolant System Activity 10/2005 through 09/2007
- Safety System Functional Failures (BWR) 4Q/2005 through 3Q/2007
- Task ID: Perform 64.713.019 ATT 17, Effluent Cumulative and Projected Dose; dated August 15, 2006 through December 12, 2007

<u>4OA2 Problem Identification and Resolution</u>

CARDs:
- CARD 07-26974: Ty-Wraps in Drywell; dated 11/03/2007
- CARD 07-27278: Human Performance Error Results in RPV Water Level Transient and Turbine Bypass Valve Stroke; dated 11/17/2007
- CARD 07-27943: Potential Emerging Trend – Ineffective/Inadequate Previous Corrective Actions; dated 12/11/2007
- CARD 07-28084: Level 2 CARDs Not Performed Correctly; dated 12/18/2007

- CARD 07-28134: Audit Finding – CARD 06-28155 Effectiveness Review Failed to Identify that the Corrective Actions were not Effective, CARD was Closed with Inadequate Corrective Actions and Documentation; dated 12/19/2007
- CARD 07-28135: Audit Finding – Untimely/dropped Corrective Actions due to Daisy-Chaining in Violation of MQA11; dated 12/19/2007

Miscellaneous:
- Observed Procedure Use & Adherence Human Performance Metrics 1, 2, 5, 6, 7
- Line Support of Training – Corrective Action Metric 14
- RCE Timeliness – Corrective Action Metric 1
- Root Cause Analysis Report, Revision 0: CARD 07-24976, NIAS Pressure Transient Event of 09/05/2007
- Root Cause Analysis Report, Revision 1: CARD 07-25631; 345 kV Disconnect Tagging Event of 10/02/2007; dated 10/31/2007
- Root Cause Analysis Report, Revision 0: CARD 07-25638; NHDP Hoist Energized with 480 Volts AC on 10/03/2007; dated 10/26/2007
- Root Cause Determination for CARD 07-25277: Investigate Effect of Opening Turbine First Floor Steam Tunnel on the Turbine Building Steam Tunnel Temperature; dated 10/29/2007

4OA3 Followup of Events and Notices of Enforcement Discretion (71153)

CARDs:
- CARD 98-12402: SIL 606: GEF-Frame Molded-Case Circuit Breaker Failure to Trip – Conversion of DER 97-0011. Sent to Records Processing 11/19/1998; dated 01/03/1997
- CARD 98-22174: Work Requests Were Completed but Not All of the Work Was Performed per the Initiating CARD; dated 11/02/1998
- CARD 01-22422: RPS Alt 'B' EPA Breaker UV light On; dated 12/16/2001
- CARD 07-26537: RPS 'B' Alternate EPA Breaker Tripped when RHR Pump 'D' Was Started; dated 10/23/2007
- CARD 07-27373: Valve Leaks, dated 11/14/2007
- CARD 07-27395: Low Level SCRAM as a Result of Reference Leg Perturbation; dated 11/15/2007

Procedures:
- 23.205, Revision 98: Residual Heat Removal System
- 23.316, Revision 36: RPA 120V AC and RPS MG Sets
- 46.000.046: Operation of the Reactor Reference Leg Backfill System; Rev. 35 dated 8/15/2006
- 29.ESP.01.2.0: Isolations and Actuations Tables, Rev 12
- NPP-23.316, Revision 18: RPS 120 VAC and RPS MG Sets

Miscellaneous:
- DTE Memo NARP-07-0111: October 11, 2007, Notice of Unusual Event Declared; 11/20/2007
- Deviation Event Report 94-0706: RPS 'B' Alternate Feed EPA Breaker Trip Failure; dated 11/22/1994
- Deviation Event Report 94-0791: Alternate EPA Breaker C7100S003F Tripped on Under Voltage; dated 12/26/1994
- Deviation Event Report 95-0784: Div 2 Alternate EPA Under Voltage Light On; dated 10/16/1995

- Deviation Event Report 97-00011: RPS Division II Under Voltage Light Lit Up; dated 01/03/1997
- Document Change Request: Procedure 23.316, RPS 120V AC and RPS MG Sets; dated 09/03/1997
- Drawing 6M721-2090 AK: System Diagram Nuclear Boiler System; dated 12/16/04
- Operator Log from 10/22/2007 to 10/25/2007
- Post SCRAM Data and Evaluation – CARD 07-27395

4OA5 Other Activities

Issue Reports:
- CARD 07/26016: Review Indication on SRV M (and others); 10/11/2007

Miscellaneous:
- Fermi 2 Event Notification Worksheet 07-00004; 10/07/2007

LIST OF ACRONYMS USED

AC	Alternating Current
ALARA	As Low As Is Reasonably Achievable
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BWROG	Boiling Water Reactor Owners' Group
CARD	Corrective Action Resolution Document
CFR	Code of Federal Regulations
CRD	Control Rod Drive
^{o}F	Degrees Fahrenheit
DRP	Division of Reactor Projects
ECCS	Emergency Core Cooling System
EECW	Emergency Equipment Cooling Water
EDG	Emergency Diesel Generator
EDP	Engineering Design Package
EOP	Emergency Operating Procedure
EPG	Emergency Procedure Guideline
HRA	High Radiation Area
IEEE	Institute of Electrical & Electronic Engineers
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
ISI	In-service Inspection
LLRT	Local Leak Rate Test
LORT	Licensed Operator Requalification Training
LPSV	Low Pressure Stop Valve
MSIV	Main Steam Isolation Valve
MSPI	Mitigating Systems Performance Index
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
PI	Performance Indicator
PMT	Post-Maintenance Testing
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
RF12	Refueling Outage 12
RHR	Residual Heat Removal
RP	Radiation Protection
RPS	Reactor Protection System
RPT	Radiation Protection Technician
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Cleanup
RWP	Radiation Work Permit
SAT	Systems Approach to Training
SDP	Significance Determination Process
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
WO	Work Order