UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, IL 60532-4352

May 2, 2011

EA-11-042

Mr. Jack M. Davis
Detroit Edison Company
Fermi 2 - 210 NOC
6400 North Dixie Highway
Newport, MI 48166

SUBJECT: FERMI POWER PLANT, UNIT 2, INTEGRATED INSPECTION REPORT 05000341/2011002

Dear Mr. Davis:

On March 31, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Fermi Power Plant, Unit 2. The enclosed report documents the results of this inspection, which were discussed on March 30, 2011, with the plant manager and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and 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, three NRC-identified findings of very low safety significance were identified. Two of these findings involved violations of NRC requirements. However, because these violations were of very low safety significance, and were entered into your corrective action program, the NRC is treating these issues as non-cited violations (NCVs) in accordance with Section 2.3.2 of the NRC Enforcement Policy. Additionally, a licensee-identified violation is listed in Section of 4OA7 of this report.

If you contest the subject or severity of this 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 Power Plant. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Fermi Power Plant.

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

Sincerely,

/RA/

John B. Giessner, Chief
Branch 4
Division of Reactor Projects

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

Enclosure: Inspection Report 05000341/2011002
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-341
License No:	NPF-43
Report No:	05000341/2011002
Licensee:	Detroit Edison Company
Facility:	Fermi Power Plant, Unit 2
Location:	Newport, MI
Dates:	January 1 through March 31, 2011
Inspectors:	R. Morris, Senior Resident Inspector R. Jones, Resident Inspector J. Bozga, Reactor Inspector, DRS T. Hartman, Resident Inspector, Perry M. Jones, Reactor Engineer M. Munir, Reactor Engineer S. Sheldon, Senior Reactor Engineer
Approved by:	J. Giessner, Chief Branch 4 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

Inspection Report (IR) 05000341/2011002, 01/01/2011 – 03/31/2011; Fermi Power Plant, Unit 2, routine integrated IR; Maintenance Effectiveness, Outage Activities, Identification and Resolution of Findings.

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Three Green findings were identified by the inspectors, two of which were considered non-cited violations (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). Cross-cutting aspects were determined using IMC 0310, "Components Within the Cross-Cutting Areas." 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 Findings

Cornerstone: Initiating Events, Barrier Integrity

- Green. A finding of very low safety significance (Green) for failure to evaluate and incorporate the operating experience received from the Boiling Water Reactors Owners Group (BWROG) Off-Gas committee was self-revealed when Fermi 2 experienced a reactor scram due to degraded condenser vacuum on October 24, 2010. The cause of the loss of vacuum was the failure of No. 3 steam jet air ejector (SJAE) steam supply to nozzle gasket, which caused steam erosion of the seating surface and loss of capacity. The licensee repaired the air ejector.

  The inspectors determined this finding was more than minor because it was associated with the equipment performance attribute of the Initiating Events Cornerstone and impacted the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The issue resulted in a scram. This finding was determined to be of very low safety significance, Green, because, while it did contribute to the likelihood of a reactor trip, it did not contribute to the likelihood that mitigating equipment would not be available. This finding was not cross-cutting because the licensee received the operating experience input over 3 years ago and was not necessarily indicative of current licensee performance. Finally, no violation of NRC requirements was identified since the SJAEs and the off-gas system are nonsafety-related. (Section 4OA2.3)

- Green. A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified by the inspectors for the failure to ensure the adequacy of the design for the reactor building crane support structure and reactor building superstructure. Specifically, the inspectors identified six representative examples where the licensee failed to perform adequate design calculations resulting in the design not being in conformance with Seismic Category I requirements as defined in Updated Final Safety Analysis Report (UFSAR) Sections 3.8.4.3.1 and 3.8.4.5.1 and referenced codes. The licensee documented the corrective actions in CARDs 10-22393, 10-22958, 10-22979, 10-23882, 10-24166, 10-26278 and 10-26691. The licensee also

performed a re-analysis of the reactor building crane support structure and reactor building superstructure to address the deficiencies, and determined the structure to be operable but nonconforming and initiated modifications.

The inspectors determined the licensee's failure to meet design requirements for Seismic Category I compliance for the reactor building crane support structure and reactor building superstructure was a performance deficiency. The performance deficiency was determined to be more than minor because the finding was associated with the Initiating Events Cornerstone attribute of design control and affected the cornerstone objective to limit the likelihood of those events that upset the plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, compliance with Seismic Category I requirements for the reactor building crane support structure and superstructure was to demonstrate safe handling of heavy loads over the reactor core, the spent fuel pool, or safety-related components. Also, the performance deficiency was determined to be more than minor because it was associated with the Barrier Integrity Cornerstone attribute of design control and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The finding screened as having very low safety significance (Green) because it was a design deficiency that did not to result in a loss of functionality/operability. The inspectors did not identify a cross-cutting aspect associated with this finding because the concern was related to calculations from the 1980s and 1990s and thus was not necessarily indicative of current licensee performance (Section 1R20).

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding and associated NCV of Technical Specification 5.4.1 for failure follow their Conduct Manual MES-43, "Instrument Calibration Specification Sheets (ICSS)," as established in Regulatory Guide 1.33, Appendix A.10, to ensure proper verification and calibration of the H_2 O_2 sample pump trip switch had been done during the annual preventative maintenance (PM) calibration. Specifically the engineering organization did not verify the actual setpoint until the inspector requested the calculations, then the licensee determined that the setpoint was out of tolerance. The licensee entered this into their corrective action program (CAP) as CARD 11-23023. The licensee completed the re-calibration of the flow switch.

  The inspectors determined that the failure to have a proper calibration of the switch was within their ability to foresee and correct, since the licensee failed to perform an evaluation when it was identified that the pump could trip at a flow setpoint in their normal band of operation established in procedures. Therefore the issue was a performance deficiency. This finding impacted the Mitigating System Cornerstone. The inspectors determined this finding was more than minor because, if left uncorrected, the early loss of the H_2O_2 sampling pump could have lead to a more significant safety concern and it was similar to the more than minor example of IMC 0612 Appendix E, 4.c. The flow switch for the H_2 O_2 sampling pump was outside of the acceptable range and would trip early causing a loss of the H_2O_2 monitoring system. This could complicate the verification of mitigating system equipment in a timely manner during plant events. The finding was determined to be of very low safety significance, Green, using IMC 0609, Significance Determination Process, Attachment 0609.04, Table 4a as all Mitigating System Cornerstone answers were 'no.' This finding has a cross-cutting aspect in the

area of Problem Identification and Resolution, Corrective Action, because Fermi 2 personnel proceeded in the face of uncertainty or unexpected circumstances by continuing with the calibration procedure and equipment use even though the pump tripped repeatedly at a setpoint value which the procedure established as acceptable, without performing an engineering evaluation that either determined the cause or provided conclusive justification for continued operation. (P.1 (c)) (Section 1R12)

B. Licensee-Identified Violations

A violation of very low safety significance that was identified by the licensee has been reviewed by inspectors. Corrective actions planned or taken by the licensee have been entered into the licensee's corrective action program. This violation and corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Fermi Unit 2 started this inspection period at 75 percent power and remained there until January 28, 2011, when the unit was taken offline for planned maintenance. The unit was restarted on February 14 and returned to 100 percent power on February 18. On March 13 the unit reduced power to 65 percent for a rod pattern adjustment and returned to 100 percent on March 14 and remained there until the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R01 Adverse Weather Protection (71111.01)

.1 Readiness for Impending Adverse Weather Condition – Heavy Snowfall Conditions

a. Inspection Scope

On February 1, 2011, a winter weather advisory was issued for expected snow squalls. The inspectors observed the licensee's preparations and planning for the significant winter weather potential. The inspectors reviewed licensee procedures and discussed potential compensatory measures with control room personnel. The inspectors focused on plant management's actions for implementing the station's procedures for ensuring adequate personnel for safe plant operation and emergency response would be available. The inspectors conducted a site walkdown including walkdowns of various plant structures and systems to check for maintenance or other apparent deficiencies that could affect system operations during the predicted significant weather. The inspectors also reviewed corrective action program (CAP) items to verify the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their CAP in accordance with station corrective action procedures. Specific documents reviewed during this inspection are listed in the Attachment to this report.

This inspection constituted one readiness for impending adverse weather condition sample as defined in inspection procedure (IP) 71111.01-05.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

- Reactor building closed cooling water (RBCCW);
- Division 1 emergency diesel generator (EDG) fuel oil transfer; and
- Reactor water cleanup (RWCU).

The inspectors selected these systems based on their risk significance relative to the Reactor Safety cornerstones 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, Updated Final Safety Analysis Report (UFSAR), Technical Specification (TS) requirements, outstanding work orders (WOs), 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 CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

These activities constituted three partial system walkdown samples as defined in IP 71111.04-05.

b. Findings

No findings were identified.

.2 Semiannual Complete System Walkdown

a. Inspection Scope

On February 8, 2011, the inspectors performed a complete system alignment inspection of the reactor building firewater 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 ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding WOs was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the CAP database to ensure system equipment alignment problems were being identified and appropriately resolved. Documents reviewed are listed in the Attachment to this report.

These activities constituted one complete system walkdown sample as defined in IP 71111.04-05.

b. Findings

No findings 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:

- Auxiliary building, sub basement, non-interruptible air system (NIAS) room;
- Reactor building, fourth floor, reactor recirculating motor generator (RRMG) area;
- Reactor building, third floor, and fourth floor standby liquid control area;
- Reactor building, second floor, north, and division 1 emergency equipment cooling water;
- Auxiliary building, relay room;
- Auxiliary building, fifth floor, division 1 control center heating, ventilation, air conditioning (CCHVAC) room; and
- Station fire truck.

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 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 that minor issues identified during the inspection were entered into the licensee's CAP. Documents reviewed are listed in the Attachment to this report.

These activities constituted seven quarterly fire protection inspection samples as defined in IP 71111.05-05.

b. Findings

No findings were identified.

1R06 Flooding (71111.06)

.1 Internal Flooding

a. Inspection Scope

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 to identify licensee commitments. The specific documents reviewed are listed in the Attachment to this report. 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 CAP 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 verify drains and sumps were clear of debris and were operable, and that the licensee complied with its commitments:

- Reactor building; southwest quad basement and sub-basement;
- Reactor building; northwest quad basement and sub-basement;
- Reactor building, northeast quad basement and sub-basement; and
- Reactor building, southeast quad basement and sub-basement.

This inspection constituted one internal flooding sample as defined in IP 71111.06-05.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On January 25, 2011, 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 usage 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. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly licensed operator requalification program sample as defined in IP 71111.11.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations (71111.12Q)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk-significant systems:

- Condensate system in (a)(1) status;
- A7100 primary containment isolation; and
- T5000 primary containment atmospheric monitoring.

In addition, the inspectors performed:

- A review of the licensee's Maintenance Rule periodic evaluation.

The inspectors reviewed events such as where ineffective equipment maintenance had resulted in valid or invalid automatic actuations of engineered safeguards systems 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 CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

This inspection constituted four quarterly maintenance effectiveness samples as defined in IP 71111.12-05.

b. Findings

Introduction: The inspectors identified a finding and associated Non-Cited Violation (NCV) of TS 5.4.1 for failure follow their Conduct Manual MES-43, "Instrument Calibration Specification Sheets," as established in Regulatory Guide 1.33, Appendix A.10, to ensure proper verification and calibration of the H_2 O_2 sample pump trip switch had been done during the annual preventative maintenance (PM) calibration. Specifically the engineering organization did not verify the actual setpoint until the inspector requested the calculations, then determined that the setpoint was out of tolerance.

Description: During a division 2 primary containment monitoring system outage, Fermi 2 experienced a condition that caused the H_2 O_2 sample pumps to trip even when the trip switch is calibrated to the center of the tolerance band (0.2 $\pm$ 0.1 scfm). The corrective action that the licensee took restored the calibration back to the as-found condition that was low within the acceptable range. No pump trips have been experienced while the sample system was in service. The sample pumps support post accident sampling requirements in the containment in accordance with Technical Requirements Manual, Section 3.3.3 (functions 6 and 7).

The licensee failed to determine the cause of the switch trips while within the tolerance of the setpoint during the PM and returned the system to service on July 2, 2010. While the licensee did perform post-maintenance testing (PMT) and determined the pumps would not trip, no formal evaluation was performed to determine why trips at the proceduralized setpoint value were being experienced.

Between early July and December 15, 2010, the exact cause of the trips remained unknown and awaited additional evaluation by the vendor and the licensee.

On December 16, 2010, Corrective Action and Resolution Document (CARD) 10-31910 was generated identifying the failed calibration of the flow meter used during the PM in which the flow switches were left at a value low within the range. It was determined the flowmeter was out of calibration low, which resulted in the adjustment of the flow switch trip setting to a value closer to normal system flow value. Taking into account that the H_2 O_2 sampling pump will trip early and possible degrade over time, the system engineer determined flow may degrade to a point where unnecessary system trips would occur prior to the next switch calibration. No specific evaluation was performed to show the system could perform its function within the acceptance criteria until the next calibration.

Upon the inspectors' questioning, engineering performed a calculation to provide reasonable assurance the pumps should remain capable of performing their intended function through the remainder of the current PM interval. But the calculation showed the instrument was outside of the acceptance criteria in the procedure.

The inspectors also noted the H_2 O_2 system was used in the emergency operating procedures and emergency action levels. If the pump tripped early, it would require manual samples to determine the H_2 and O_2 concentration inside of the drywell. The samples would take 45 minutes or longer to perform an analysis, thus delaying information used by operations and emergency response personnel to make decisions during an emergency.

Analysis: The inspectors determined that the failure to have a proper calibration of the switch was within their ability to foresee and correct, since the licensee failed to perform an evaluation when it was identified that the pump could trip at a flow setpoint in their normal band of operation established in procedures. Therefore the issue was a performance deficiency. This finding impacted the Mitigating System Cornerstone. The inspectors determined this finding was more than minor because if left uncorrected the early loss of the H_2O_2 sampling pump could have lead to a more significant safety concern and it was similar to IMC 0612 Appendix E, 4.c. The flow switch for the H_2 O_2 sampling pump was outside of the acceptable range and would trip early causing a loss of the H_2O_2 monitoring system. This could complicate the verification of mitigating system equipment in a timely manner during plant events. While it did not contribute to the likelihood of a loss of the mitigating equipment, it did contribute to the likelihood that verification of proper operation of mitigating equipment in a timely manner for emergency events would be delayed while performing a manual containment air sample. The finding was determined to be of very low safety significance, Green, using IMC 0609, Significance Determination Process, Attachment 0609.04, Table 4a as all Mitigating System Cornerstone answers were 'no.' This finding has a cross-cutting aspect in the area of Problem Identification and Resolution, Corrective Action, because Fermi 2 personnel proceeded in the face of uncertainty or unexpected circumstances by continuing with the calibration procedure and equipment use, even though the pump tripped repeatedly at a setpoint value which the procedure established as acceptable without performing an engineering evaluation that either determined the cause or provided conclusive justification for continued operation. (P.1 (c))

Enforcement: Technical Specification 5.4.1 requires written procedures shall be established, implemented, and maintained covering Regulatory Guide 1.33, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A, Section 10, "Chemical and Radiological Control Procedures," recommends procedures for control of measuring and testing of equipment used to determine concentration and species of radioactivity in liquids and gases prior to release, including representative sampling, validity of calibration techniques, and adequate analysis. MES41, Revision 4, Step 3.3.4 states in part, "…the setpoints content of ICSS are controlled and require a Design Change Document to revise."

Contrary to the above, on December 16, 2010, the licensee failed to have an adequate calibration with proper setpoints for the H_2O_2 sampling pump; and did not evaluate the condition to determine the as-left setpoint. The incorrect switch setpoint could prevent the H_2O_2 monitoring system from performing its function during accident conditions as

specified in Technical Requirements Manual 3.3.3 Function 6 and 7. Because this violation was of very low safety significance (Green), was not repetitive or willful, and it was entered into the licensee's CAP as CARD 11-23023, "NRC Concern: T50N105B Flow Switch Calibration Acceptance," this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000345/2011002-01; Failure to Fully Evaluate the Failure of H_2 O_2 Sampling Pump Trips during Calibration).

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 prior to removing equipment for work:

- Risk during the transition from power operations work to a forced outage;
- Risk during the transition from forced outage work to normal operations;
- Risk during NIAS and high pressure coolant injection (HPCI) maintenance;
- Risk during Division 2 residual heat removal (RHR)/residual heat removal service water (RHRSW) safety system outage (SSO), downpower for minor rod pattern adjustment (RPA) to achieve 100 percent rod line, major RPA; and
- Risk during division 1 CCHVAC SSO, and dedicated shutdown operability.

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 maintenance risk assessments and emergent work control activities constituted five samples as defined in IP 71111.13-05.

b. Findings

No findings were identified.

1R15 Operability Evaluations (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- CARD 10-31926, Turbine building HVAC south exhaust fan bearing replacement;
- CARDs 10-32197, 11-21153, and 11-21136, Drywell insulation type installed in localized recirculating piping areas, calculation errors, and heat shields found above RRMG 'A';
- Intermediate Range Monitor (IRM) F and IRM D issue from startup in 2010 (MES27 Enclosure A, page 14);
- CARD 11-22407, T2300F450A torus-to-reactor building vacuum breaker did not indicate properly during surveillance testing;
- CARD 10-31614, High hydrogen seal oil temperature at hydrogen split ring end; and
- CARDs 11-22500, 11-22541; ODMI 11-004; Spurious AVR general alarm 4D53.

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. Additionally, the inspectors reviewed a sampling of corrective action documents to verify the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment to this report.

This operability inspection constituted six samples as defined in IP 71111.15-05.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

.1 Plant Modifications

a. Inspection Scope

The inspectors reviewed the following modification:

- Scaffolding installed during Refueling Outage 14 for use during the cycle.

The inspectors reviewed the configuration changes and associated 10 CFR 50.59 safety evaluation screening against the design basis, the UFSAR, and the TS, as applicable, to verify the modification did not affect the operability or availability of the affected system. The inspectors, as applicable, observed ongoing and completed work activities to ensure the modifications were installed as directed and consistent with the design control documents; the modifications operated as expected; post-modification testing adequately demonstrated continued system operability, availability, and reliability; and operation of the modifications did not impact the operability of any interfacing systems. As applicable, the inspectors verified relevant procedure, design, and licensing documents were properly updated. Lastly, the inspectors discussed the plant modification with operations, engineering, and training personnel to ensure the individuals were aware of how the operation with the plant modification in place could impact overall plant performance. Documents reviewed in the course of this inspection are listed in the Attachment to this report.

This inspection constituted one temporary plant modification sample as defined in IP 71111.18-05.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

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

- WO 32145027, North Condensate Pump Repair and PMT;
- RRMG set coupling work (PMT);
- High pressure turbine control valve (HPCV) 1 removal and repair, PMT bypass valve 24.109.02 and HPCV/SV No. 1 thru No. 4 PMT 24.110.05;
- WOs 30278677, 30278577, 30278671, 31611700, and 29898593 – NIAS Maintenance PMTs;
- HPCI run following motor operated valve maintenance;
- Engineering Design Package (EDP)-36238 Leading Edge Flow Monitor; and
- WO 31052217, Replace Feedback Potentiometer in South RRMG Set Tube Positioner.

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 TSs, the UFSAR, 10 CFR Part 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 CAP and the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the Attachment to this report.

This inspection constituted seven PMT samples as defined in IP 71111.19-05.

b. Findings

No findings were identified.

1R20 Outage Activities (71111.20)

.1 (Closed) Unresolved Item 05000341/2010011-01: Reactor Building Crane and Reactor Building Crane Support Structure/Superstructure Design Calculation Issue

a. Inspection Scope

The inspectors reviewed additional information provided by the licensee pertaining to Unresolved Item (URI) 05000341/2010011-01. Specific documents reviewed during the inspection are listed in the Attachment to this report.

b. Findings

Introduction: A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified by the inspectors for the failure to ensure the adequacy of the design for the reactor building crane support structure and superstructure design. Specifically the building design did not comply with Seismic Category I requirements as referenced in UFSAR Sections 3.8.4.3.1 and 3.8.4.5.1 and required codes.

Description: The reactor building crane support structure and superstructure are required to be Seismic Category I per UFSAR, Table 3.2-1, "Structures, Systems, and Components Classification." The reactor building structure provides secondary containment when the primary containment is closed and in service, and it provides primary containment during reactor refueling and maintenance operations when the primary containment is open as described in UFSAR 3.8.4.1.1.1. The reactor building structure also supports the reactor building crane, which handles heavy loads such as the reactor and drywell heads, reactor shield plugs, fuel bundles or a spent fuel cask over the reactor core, over the spent fuel pool, or over safety-related components during power operations as well as shutdown operations.

During a review of calculations for the reactor building crane support structure and reactor building superstructure, the inspectors identified the following six representative examples in which the licensee failed to meet the design requirements:

- Calculation No. 4.02.04, “Superstructure Roof Framing Bracing System,” Volume I, Revision A, and Calculation No. 4.02.09, “Reactor Building Superstructure Steel Girt and Column Framing Design,” Volume I, Revision A: UFSAR, Section 3.8.4.5.1 states, “Stresses and strains in the structural steel used for the reactor/auxiliary building are limited to those specified in the 1969 American Institute of Steel Construction (AISC) Specifications….” The requirement in the AISC was that the allowable stress was based on the specified minimum yield stress of the material. The licensee used certified material test reports or actual material yield strength for the evaluation of the reactor building horizontal and vertical bracing members, girts, crane runway girder, building columns, and column anchor plate. In one instance, the licensee used 0.7 times the actual ultimate material tensile strength to calculate allowable stresses for the reactor building vertical bracing member. The use of actual material yield strength and 0.7 times the actual ultimate tensile strength did not meet AISC requirements. The licensee documented these deficiencies in CARDs 10-22393, 10-22979 and 10-24166.

- Calculation No. 4.02.04, “Superstructure Roof Framing Bracing System,” Volume I, Revision A: UFSAR, Section 3.8.4.5.1 states, “Stresses and strains in the structural steel used for the reactor/auxiliary building are limited to those specified in the 1969 AISC specifications….” In addition to the use of actual material strength for allowable stress, the reactor building horizontal bracing member evaluations did not satisfy the AISC requirements of Section 1.6.1 and Section 1.8.2. Specifically, AISC Section 1.6.1 requires that Cm, which is a coefficient, shall be taken as ‘one’ if the ends of the compression member are unrestrained against rotation. American Institute of Steel Construction Section 1.8.2 requires the effective length factor for a compression member to be unity or one. The reactor building horizontal bracing member analysis did not meet the requirements of AISC Section 1.6.1 and 1.8.2 in that the licensee did not use a Cm value and effective length factor of one. The licensee documented this deficiency in CARD 10-22979.

- Calculation No. 4.02.04, “Superstructure Roof Framing Bracing System,” Volume I, Revision A: UFSAR, Section 3.8.4.5.1 states, “Stresses and strains in the structural steel used for the reactor/auxiliary building are limited to those specified in the 1969 AISC Specifications, Part I, when the loading combinations listed in Tables 3.8-18 and 3.8-19 were being designed for.” In addition to not meeting the minimum yield strength AISC requirements for allowable stress and AISC Section 1.6.1 and Section 1.8.2 requirements, the reactor building vertical bracing member evaluations used a dynamic increase factor to increase the allowable acceptance limits for the severe and extreme environmental load combination. The reactor building vertical bracing member evaluations did not satisfy the requirements of UFSAR Table 3.8-18 and UFSAR Table 3.8-19, as well as AISC requirements. The licensee documented this deficiency in CARD 10-22979.

- Calculation No. DC-6019, “Assessment of the Interior Columns for the Reactor Building Steel Superstructure Including Crane Lifted Load”; Volume IA, Revision 0: UFSAR, Section 3.8.4.5.1 states, “Stresses and strains in the

structural steel used for the reactor/auxiliary building are limited to those specified in the 1969 AISC Specifications, Part I, when the loading combinations listed in Tables 3.8-18 and 3.8-19 were being designed for." Updated Final Safety Analysis Report Table 3.8-18, "Loading Combinations for Steel Structures Elastic Design," requires live load to be applied in combination with dead load, crane lifted load, and seismic load for the severe and extreme environmental load combination. The licensee's reactor building column and reactor building column anchorage analysis did not address the live load condition. In addition, dead load supported by the reactor building columns such as the crane bottom block and wire rope, platform along column row F, girts, standby gas treatment supports attached to column F13 and F15, HVAC supports along column row A and pipe supports T46-5909-G01 and T26-5909-G02 were not addressed in the reactor building column and reactor building column anchorage analysis as well. The licensee documented these deficiencies in CARDs 10-22393 and 10-26278.

- Calculation No. 4.02.01, "Crane Girder Splice," Section 12, Revision 0, and Calculation No. 4.02.04, "Superstructure Roof Framing Bracing System," Volume I, Revision A: UFSAR, Section 3.8.4.5.1 states, "Stresses and strains in the structural steel used for the reactor/auxiliary building are limited to those specified in the 1969 AISC Specifications, Part I, when the loading combinations listed in Tables 3.8-18 and 3.8-19 were being designed for." Updated Final Safety Analysis Report Table 3.8-18, "Loading Combinations for Steel Structures Elastic Design" requires crane lifted load to be applied in combination with dead load, live load, and seismic load for the severe and extreme environmental load combination. In addition to not meeting the minimum yield strength AISC requirements for allowable stress, the licensee used a dynamic increase factor to increase the acceptance limits for the severe and extreme environmental load combination. The licensee's reactor building crane girder analysis did not address the crane lifted load condition for the severe and extreme environmental load combination and did not use acceptance criteria that was in accordance with design and licensing basis requirements. Also, the crane girder runway splice connection did not address the severe and extreme environmental load combination in accordance with the UFSAR Table 3.8-18 requirements. The licensee documented these deficiencies in CARDs 10-22958 and 10-26691.

- The licensee did not evaluate the crane runway girder rails and the rail clip connections for the normal, severe, and extreme environmental load combination in accordance with the UFSAR Table 3.8-18 requirements. After inspection of the crane calculation and the crane runway girder calculation, the inspectors identified that the crane analysis transferred the dead loads, crane lifted load and seismic loads to the crane runway girder through the crane runway girder rails and rail clip connections to the crane support structure analysis. The crane runway girder rails and the rail clip connections were not evaluated for such loads. The licensee documented this deficiency in CARD 10-23882.

As a result of the inspectors' concerns, the licensee performed an operability evaluation and reanalysis of the reactor building crane support structure and reactor building superstructure to address the various deficiencies, including those above, determined the structure to be operable but nonconforming, and initiated modifications. As

documented in 1R15 of Fermi IR 2010005, the inspectors reviewed the operability evaluation and did not identify any findings of significance.

Analysis: The inspectors determined the licensee's failure to meet design requirements for Seismic Category I compliance for the reactor building crane support structure and reactor building superstructure was a performance deficiency. The performance deficiency was determined to be more than minor because the finding was associated with the Initiating Events Cornerstone attribute of design control and affected the cornerstone objective to limit the likelihood of those events that upset the plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, compliance with Seismic Category I requirements for the reactor building crane support structure and superstructure was to demonstrate safe handling of heavy loads over the reactor core, the spent fuel pool, or safety-related components. Also, the performance deficiency was determined to be more than minor because it was associated with the Barrier Integrity Cornerstone attribute of design control and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, compliance with Seismic Category I design basis requirements was to ensure the reactor building superstructure would function as required during a Seismic Category I design basis event and not adversely affect the secondary containment.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1--Initial Screening and Characterization of Findings," Table 4a, for the Initiating Events Cornerstone. The inspectors answered "no" to all the questions in the Initiating Events column based on the licensee determining the reactor building crane support structure and superstructure were operable but nonconforming and concluded the finding was of very low safety significance (Green). The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1--Initial Screening and Characterization of Findings," Table 4a, for the Barrier Integrity Cornerstone. The finding screened as having very low safety significance (Green) because it was a design deficiency of the physical integrity of the reactor containment. The inspectors answered "no" to all the questions in the Containment Barrier column based on the licensee determining that the reactor building crane support structure and superstructure were operable but nonconforming and concluded that the finding was of very low safety significance (Green).

The inspectors did not identify a cross-cutting aspect associated with this finding because the concern was related to calculations from the 1980s and 1990s and thus was not necessarily indicative of current licensee performance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," states in part, that the design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.

Contrary to the above, on March 19, 2010, through January 21, 2011, the inspectors determined that for Calculation No. 4.02.01, Section 12, Revision 0; Calculation

No. 4.02.04, Volume I, Revision A; Calculation No. 4.02.09, Volume I, Revision A; and Calculation No. DC-6019, Volume IA, Revision 0, the licensee's design control measures failed to ensure adequacy of the design. Specifically, these calculations did not conform to and were nonconservative with respect to UFSAR and Code requirements.

Because this violation was of very low safety significance (Green) and it was entered into the licensee's CAP as CARDs 10-22393, 10-22958, 10-22979, 10-23882, 10-24166, 10-26278 and 10-26691, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000341/2011002-02: Design Control Measures Failed to Ensure Adequacy of the Design Relating to the Reactor Building Crane Support Structure and Reactor Building Superstructure).

This NRC-identified violation closes URI 05000341/2010011-01, "Reactor Building Crane and Reactor Building Crane Support Structure/Superstructure Design Calculation Issue".

.2 Other Outage Activities

a. Inspection Scope

The inspectors evaluated outage activities for a scheduled outage that began on January 28, 2011, and continued through February 14, 2011. The inspectors reviewed activities to ensure the licensee considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed or reviewed the reactor shutdown and cooldown, outage equipment configuration and risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, control of containment activities, startup and heatup activities, and identification and resolution of problems associated with the outage activities listed below. Documents reviewed during the inspection are listed in the Attachment to this report.

- Licensee configuration management, including maintenance of defense-in-depth commensurate with the OSP 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, accounting for instrument error.
- Monitoring of decay heat removal processes, systems, and components.
- 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.
- 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 emergency core cooling system (ECCS) suction strainers, and reactor physics testing.
- Licensee identification and resolution of problems related to outage activities.

This inspection constituted one other outage sample as defined in IP 71111.20-05.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

.1 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:

- Procedure 44.010.109, Division 2 IRM Calibration;
- Procedure 44.030.001, ECCS Core Spray System, Division 1, Logic Functional Test (in service test);
- Procedure 24.425.02, Drywell Personnel Hatch Weekly Functional Test;
- Procedure 24.307.16, EDG 13 Start and Load Test;
- Procedure 24.307.17, EDG 14 Start and Load Test;
- WO 32354416, Perform Control Rod Cycling Prior to Startup from Planned Outages; and
- Procedure 44.030.056, ECCS Recirculation Riser Differential Pressure, Division 1 Functional Test.

The inspectors observed in-plant activities and reviewed procedures and associated records to determine the following:

- did preconditioning occur;
- were the effects of the testing adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- were acceptance criteria clearly stated, demonstrated operational readiness, and consistent with the system design basis;
- plant equipment calibration was correct, accurate, and properly documented;
- as-left setpoints were within required ranges; and 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 inservice testing activities, testing was performed in accordance with the applicable version of Section XI, American Society of Mechanical Engineers 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 CAP.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted six routine surveillance testing samples and one inservice testing sample, as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings were identified.

1EP6 Drill Evaluation (71114.06)

.1 Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of a routine licensee emergency drill on March 15, 2011, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the simulator control room and technical support center to determine whether the event classification, notifications, and protective action recommendations were performed in accordance with procedures. The inspectors also attended the licensee drill critique to compare any inspector-observed weakness with those identified by the licensee staff in order to evaluate the critique and to verify whether the licensee staff was properly identifying weaknesses and entering them into the CAP. As part of the inspection, the inspectors reviewed the drill package and other documents listed in the Attachment to this report.

This emergency preparedness drill inspection constituted one sample as defined in IP 71114.06-05.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

Cornerstones: Initiating Events

4OA1 Performance Indicator Verification (71151)

.1 Unplanned Scrams per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams per 7000 Critical Hours performance indicator (PI) for the period from the fourth quarter 2009 until the fourth quarter 2010. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports, and NRC inspection reports for the period of October 2009 through December 2010 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one unplanned scrams per 7000 critical hours sample as defined in IP 71151-05.

b. Findings

No findings were identified.

.2 Unplanned Scrams with Complications

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams with Complications PI for the period from the fourth quarter 2009 until the fourth quarter 2010. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports, and NRC integrated inspection reports for the period of October 2009 through December 2010 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one unplanned scrams with complications sample as defined in IP 71151-05.

b. Findings

No findings were identified.

.3 Unplanned Transients per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Transients per 7000 Critical Hours PI for the period from the fourth quarter 2009 until the fourth quarter 2010. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, maintenance rule records, event reports, and NRC integrated inspection reports for the period of October 2009 through December 2010 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one unplanned transients per 7000 critical hours sample as defined in IP 71151-05.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

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 CAP at an appropriate threshold, adequate attention was being given to timely corrective actions, and adverse trends were identified and addressed. Attributes reviewed included: identification of the problem was complete and accurate; timeliness was commensurate with the safety significance; 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 the classification, prioritization, focus, and timeliness of corrective actions were commensurate with safety and sufficient to prevent recurrence of the issue. Minor issues entered into the licensee's CAP as a result of the inspectors' observations are included in the Attachment to this report.

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 Section 1 of this report.

b. Findings

No findings 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 followup, the inspectors performed a daily screening of items entered into the licensee's CAP. 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 were identified.

.3 Selected Issue Followup Inspection: Reactor Scram due to Loss of Vacuum - Root Cause

a. Inspection Scope

The inspectors selected the reactor scram due to loss of vacuum for an in-depth review. This event, which occurred on October 24, 2010, resulted from a main turbine trip. Condenser vacuum had degraded actuating a relay that initiated a turbine control valve fast closure, which initiated the reactor scram. Cause of the degraded condenser vacuum was erosion of the No. 3 steam jet air ejector steam supply first stage nozzle. This resulted in a loss of air ejector capacity. The inspectors reviewed CARD 10-29450 the root cause evaluation, and the actions taken to address the causes of the steam jet air ejector erosion. Documents reviewed in this inspection are listed in the Attachment.

This inspection constituted one in-depth problem identification and resolution sample as defined in IP 71152-05.

b. Findings.

Introduction: A finding of very low safety significance (Green) for failure to evaluate and incorporate the operating experience received from the Boiling Water Reactors Owners Group (BWROG) Off-Gas committee was self-revealed when Fermi 2 experienced a reactor scram due to degraded condenser vacuum on October 24, 2010. The cause of the loss of vacuum was the failure of No. 3 steam jet air ejector (SJAE) steam supply to nozzle gasket, which caused steam erosion of the seating surface and loss of capacity.

Description: Just prior to reducing power for Refueling Outage 14, at 1641 hours on October 24, 2010, Fermi 2 experienced an automatic scram due to degraded main condenser vacuum. The licensee determined the cause of the loss of vacuum was the failure of No. 3 SJAE steam supply to nozzle gasket, which caused steam erosion of the seating surface and loss of capacity.

The licensee determined the root cause of the failure was that SJAE degradation was not determined through PM. Also, the contributing cause was determined to be that the failure of multiple instruments to alarm and actuate at specified set points limited the amount of time for operators to respond to degrading vacuum.

In approximately 2001-2002 as part of an integrated PM program, the licensee identified SJAEs as critical components and determined the need to develop a maintenance strategy. In September 2004, PMs were created for the SJAEs; however, there was no requirement included to disassemble or inspect the nozzle to steam supply joint. A system engineer did review the SJAE inspection PM requirements and determined they were adequate based upon the information discussed during a BWROG session attended.

Further, the inspectors determined that in May 2007, a BWROG technical paper, Generic BWR Off-Gas SJAE Maintenance and Inspection, was provided to the licensee, which included details of erosion of the nozzle to steam supply joint due to gasket leakage at Browns Ferry. The licensee never incorporated this operating experience at Browns Ferry into their CAP, nor into a revision of the PM inspection requirements for SJAEs. Subsequent to receiving this Browns Ferry operating experience, the licensee inspected No. 4 SJAE in October 2007 and No. 2 SJAE in April 2009; however, these inspections did not include disassembly and inspection of the nozzle to steam supply joint. The inspectors determined that the licensee failed to comply with Procedure MLS04, Operating Experience Program, Revision 17, which provided guidance for determining potential applicability in evaluating operating experience, specifically “if there is a causal relationship, the issue is applicable.” The license did not recognize the causal link in the Browns Ferry experience, and failed to enter this OE into the CARD.

Analysis: The inspectors determined the failure to incorporate the operating experience received from the BWROG Off-Gas committee regarding the Browns Ferry erosion of the nozzle to steam supply joint was a performance deficiency that required an SDP evaluation. The inspectors determined this finding was more than minor because it was associated with the equipment performance attribute of the Initiating Events Cornerstone and impacted the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, this issue resulted in a plant scram. This finding was determined to be of very low safety significance, Green, because, while it did contribute to the likelihood of a reactor trip, it did not contribute to the likelihood that mitigating equipment would not be available, since the other SJAE No. 4 remained in service following the scram, and no safety related functions were impacted. This finding was not cross-cutting because the licensee received the operating experience input over 3 years ago.

Enforcement: No violation of NRC requirements was identified since the SJAEs and the off-gas system are nonsafety-related. (FIN 05000341/2011002-03: Reactor Scram due to Loss of Vacuum).

.4 Selected Issue Followup Inspection: Missed Opportunity Review for Component Design Basis Inspection Results

a. Inspection Scope

The inspectors selected the following CARD for an in-depth review:

- CARD 10-26632, Missed Opportunity Review for Component Design Basis Inspection (CDBI) Results.

The inspectors reviewed the missed opportunity review and inspected the following attributes during their review of the above evaluation:

- complete and accurate identification of the problem in a timely manner commensurate with its safety significance and ease of discovery;
- systematic methodology was used to evaluate the problem and identify the root and contributing causes;
- consideration of the extent of condition, generic implications, common cause and previous occurrences;
- classification and prioritization of the resolution of the problem, commensurate with safety significance;
- identification of the root and contributing causes of the problem; and
- identification of corrective actions, which were specified for each root and contributing cause.

This inspection constituted completion of one in-depth problem identification and resolution sample as defined in IP 71152-05

b. Observations

The CDBI team completed an inspection (IR 05000341/2010-006) identifying weaknesses in key electrical engineering design calculations, weaknesses in operating experience, and a maintenance rule scope deficiency for the turbine building HVAC system. A cross-cutting aspect of human performance, resources, complete and accurate design document (H.2(c)) was assigned to the five electrical engineering design calculation issues. The initial apparent cause evaluation by the licensee (CARD 10-20823) identified that process weakness only existed within the electrical engineering discipline. This caused the inspectors to prompt the licensee to reconsider the other engineering disciplines, since the Independent Spent Fuel Storage Installation (ISFSI) inspectors had already identified weaknesses in civil/structural engineering design calculations. In response to these CDBI inspection findings and the ISFSI civil/structural engineering design calculation questions, the licensee prepared a comprehensive Engineering Human Performance Excellence Plan and also chartered a root cause team in August 2010 to conduct a formal root cause evaluation (RCE) under CARD 10-26632, originally titled, Root Cause Evaluation for 2010 Component Design Basis Inspection Results.

After 6 months of effort, the inspectors were advised that the team leader of the RCE for CDBI Results (CARD 10-26632) was being changed as of January 18, 2011. Subsequently, the CARD 10-26632 title was also revised to, "Missed Opportunity Review for Component Design Basis Inspection Results." The inspectors noted that the missed opportunity review was not a formal RCE performed with the scope outlined in the RCE team charter. Advised by the management sponsor that the missed opportunity review was completed, the inspectors provide the following observations on CARD 10-26632:

1. CARD 10-26632 was never completed and issued. A formal RCE of the CDBI findings was not conducted. There was no systematic evaluation of the multiple issues documented, and likewise, no root, contributing, or common causes that were identified. This card was revised to require a missed opportunity review. However, a missed opportunity review is not a systematic causal evaluation of the multiple issues documented. Rather, the missed opportunity review scope clarified that it was evaluating three target barriers. The card provided an RCE team charter, which was still in effect following the revision of the card to provide a missed opportunity review.

2. MQA12, "Root Cause Evaluations," Section 4.3.4, regarding analyses, step 3 advises, "If the picture is not complete and cannot be further developed, communicate this issue with the management sponsor and document the basis for ending the investigation and analysis in the RCE section of the report." Documenting the original RCE team findings and the basis for terminating the (initial team's) investigation was not done for this CARD. Likewise, MQA12, Section 4.6.2 regarding the RCE report, advises, "It must be thorough and provide enough detail for an independent third party reviewer to follow and understand." Again, this was not done for the original RCE report.

   The history of CARD 10-26632 identified a potential programmatic weakness, in that the management sponsor did not approve the RCE report prepared by the chartered RCE team, changed the team membership, and revised the level 2 card from a formal RCE to a missed opportunity review (which is not a formal RCE). The CARD Review Board approved the revision of this CARD to level 2. The evaluation effort performed over the period from initiation on August 4, 2010, until revision of the team on January 18, 2011, was never issued or documented in the CARD. There is no record of the bases for the changes or evaluation of the effort done by the original, more independent root cause team.

3. This CARD was a level 2 CARD based upon MQA11, "Condition Assessment Resolution Document," Enclosure B, Level 2, item 14, management discretion and required an RCE. Since this card is not yet closed the inspectors will followup on this in the future.

Although the missed opportunity review was completed, the requirements for RCEs provided by QA Conduct Manual MQA12 were not done for CARD 10-26632. Specifically the RCE for CARD 10-26632 was not completed or closed as required in a timely manner. Further, review of MQA11 and MQA12 identified the evaluation performed and the revision of CARD 10-26632 title, scope of evaluation, team membership, and documentation of the original analysis effort did not comply with

QA Conduct Manual requirements for CARDs and RCEs. The licensee wrote CARD 11-24268 to address the issue.

c. Findings

No findings of significance were identified.

.5 Selected Issue Followup Inspection: EDP 3521, Backfit Modification for Compliance with NRC BTP PSB-1 Position B.1.b.1; Revision A

a. Inspection Scope

The inspectors reviewed backfit modification, associated calculations, drawings, work orders, and CARDS that documented the various issues/deficiencies identified by the NRC during the last CDBI at Fermi. In particular, the inspectors performed a very intensive review of the revised modification and the associated calculations to ascertain whether or not the issues identified during the CDBI were adequately addressed by the licensee. Upon review of the auxiliary power system analysis and undervoltage relay setpoints calculations, the inspectors determined the licensee had adequately addressed the issues of spurious grid separation, motor starting voltage, degraded voltage relay setpoint and time delay; deadband issue of the load tap changer of transformer 64, and voltage drop due to loss of the unit. The inspectors also verified the licensee appropriately modified the EDG start and breaker trip scheme and the load shed scheme in order to bring it into compliance with Branch Technical Position PSB-1. The inspectors also reviewed a sampling of work orders to ensure the degraded voltage relays were set according to the design calculations.

This inspection constituted completion of one in-depth problem identification and resolution sample as defined in IP 71152-05.

b. Observations

During the review of the calculations, especially the power system analysis ETAP calculation, the inspectors had some difficulty in piecing together the information because of the complexity and size of the calculation; however, discussion with the licensee and their Architect Engineer clarified the issues. Presently, the licensee is going through a calculation reconstitution effort and as such the inspectors believe this will help in further enhancing the quality of the calculations. In conclusion, the inspectors verified the modifications associated with the backfit were entered into the appropriate design calculations accurately.

c. Findings

No findings were identified.

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

.1 North Reactor Feed Pump Oscillation and Trip

a. Inspection Scope

The inspectors reviewed the plant's response to the manual tripping of the north reactor feed pump (NRFP) due to observing speed oscillations during power reduction for the rod pattern adjustments. The apparent cause was failure of the system 1 RFP turbine control system. The system was isolated and system 2 RFP turbine control system was placed in service with tachometer 'A' in service. When NRFP speed was raised, oscillations were again observed. However, plant control traces were stable. Tachometer 'B' was placed in service and the NRFP speed was restored. There was no impact on safety related equipment and no loss of mitigating system capability occurred. Documents reviewed in this inspection are listed in the Attachment to this report.

This event followup review constituted one sample as defined in IP 71153-05.

b. Findings

No findings were identified.

.2 (Closed) Licensee Event Report 05000341/2010-003-00: Automatic Reactor Scram due to Degraded Condenser Vacuum

a. Inspection Scope

This event, which occurred on October 24, 2010, resulted from a main turbine trip. Condenser vacuum had degraded actuating a relay that initiated a turbine control valve fast closure, which initiated the reactor scram. All control rods fully inserted except for control rod 10-35, which stopped moving at position 42. Control rod 10-35 was then manually inserted to the full in position. Cause of the degraded condenser vacuum was erosion of the No. 3 steam jet air ejector steam supply first stage nozzle. This resulted in a loss of air ejector capacity. The inspectors reviewed and observed the actions taken to address control rod 10-35 in Inspection Report 05000341/2010-005. The evaluation of the actions and causes of the steam jet air ejector erosion were reported in Section 4OA2 of this report as a selected issue followup inspection. Documents reviewed in this inspection are listed in the Attachment. This Licensee Event Report (LER) is closed.

b. Findings

See Section 4OA2.3, Selected Issue for Followup Inspection, for findings identified.

4. OTHER ACTIVITIES

4OA5 Other Activities

.1 (Closed) Design Basis Change for the Ultimate Heat Sink (Unresolved Item 05000341/2007003-04)

On May 23, 1995, the licensee completed 10 CFR 50.59 Safety Evaluation (SE) 95 0017, which changed the period of time that the ultimate heat sink (UHS) needed to provide sufficient cooling capacity to the safety-related service water pumps without makeup. Before the change, the UHS was required to provide 30 days of water without makeup to the safety-related service water pumps, which includes the RHRSW pumps, the emergency equipment service water pumps, and the diesel generator service water pumps. This safety evaluation changed this period of time requirement from 30 days to 7 days with subsequent makeup. As a result of this evaluation, the licensee determined the change did not involve an unreviewed safety question; and therefore, it did not require prior NRC approval. During a CDBI documented in Inspection Report 05000341/2007003, the inspectors questioned the adequacy of a 10 CFR 50.59 safety evaluation and the licensee's basis for determining that revisions to UFSAR Section 9.2.5, "Ultimate Heat Sink," did not require prior NRC approval before implementation. This issue was considered unresolved pending further NRC review of Fermi's licensing basis for the period of time required for the UHS to provide sufficient cooling capability to the safety-related service water pumps and to determine if the licensee had adequately analyzed whether the change involved an unreviewed safety question. The inspectors were specifically concerned due to changes to the Fermi TS bases.

During this inspection, the inspectors reviewed the 10 CFR 50.59 SE for UFSAR changes that the licensee had implemented and consulted with the TS Branch in NRR. The inspectors determined this was not a change to the TSs. However, the inspectors also noted that at the time of the change, 10 CFR 50.59(2)(i) stated, in part, "…a proposed change, test, or experiment shall be deemed to involve an unreviewed safety question if the probability of…malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased. As the licensee incorporated reliance on a series of manual actions to provide a suction source to the safety-related service water pumps, their probability of failure to complete their design function increased by the probability of failure to complete these manual actions and makeup to the UHS. Therefore, this involved an unreviewed safety question and the licensee should have submitted the change for prior approval at the time it was evaluated.

As 10 CFR 50.59 has been revised, current NRC policy requires a review of old issues against the new rule. The inspectors reviewed the guidance in Revision 1 of NEI 96-07, which was endorsed by the NRC in Regulatory Guide 1.187. The inspectors determined that NEI 96-07, Section 4.3.2, Example 4, was applicable to the licensee's particular circumstances in that the change involved a new or modified operator action that supported a design function credited in safety analyses. The inspectors verified the licensee had incorporated the actions in plant procedures, operator training programs had demonstrated the action could be completed in the time required and alternatives had been evaluated should errors occur in the performance of manual actions. This

would justify the increased probability of malfunction is minimal and, therefore, would not require prior approval under the current rule.

In accordance with Section 3.5 of the Enforcement Policy, and after consultation with the Director, Office of Enforcement, the NRC has chosen to exercise enforcement discretion and not cite a violation for this issue. EA 11-042 is the reference tracking document. This URI is closed with no further action required.

4OA6 Management Meetings

.1 Exit Meeting Summary

On March 30, 2011, the inspectors presented the inspection results to T. Conner, Plant Manager, and other members of the licensee staff. The licensee acknowledged the issues presented.

.2 Interim Exit Meetings

Interim exits were conducted for:

- The closure of URI 05000341/2010011-01 as an NCV with the Nuclear Licensing Manager, Mr. Rodney Johnson, and other members of the licensee's staff via telephone on January 21, 2011. Licensee personnel acknowledged the inspection results inspected.

- The resolution of URI 05000341/2007003-04 on March 24, 2011, with Mr. Rodney Johnson and other members of the licensee staff.

The inspectors confirmed that none of the potential report input discussed was considered proprietary.

4OA7 Licensee-Identified Violations

The following violation of very low significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section 2.3.2 of the NRC Enforcement Policy for being dispositioned as an NCV.

> During Refueling Outage 14 in 2010 the licensee performed a walkdown to inventory and validate the debris source term inside the primary containment. The licensee identified drywell insulation in recirculation piping areas that was not the appropriate material for use in that area of the drywell. The types of insulation in the areas potentially affected by a loss-of-coolant accident were evaluated during the ECCS strainer replacement project and associated EDPs in 1998. The min-K insulation was to be replaced with a more acceptable type (NUKON) during the implementation of the EDPs. Ten of the locations containing the min-K insulation were identified following the outage. The licensee indentified the insulation for replacement at the next available drywell opening. The replacement occurred during the January 2011 planned outage. Appendix B, Criteria III of 10 CFR 50, "Design Control," states in part "The design control measures shall provide for verifying or checking the adequacy of

design…." Contrary to the above, from 1998 and prior to January 2011,the licensee did not maintain accurate design information and failed to identify these sections of insulation for replacement in 1998. Because the amount of insulation was small and within the existing available design margin for the ECCS suction strainer debris source term (DC-5979), the strainer was still operable; and the finding was determined to be of very low safety significance. The licensee entered the issue into their corrective action program as CARD 10-32197, "Confirmation of insulation type installed in localized recirulation piping areas within the drywell." The finding was determined to be a licensee-identified NCV of 10 CFR 50 Appendix B, Criteria III.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

T. Conner, Plant Manager
C. Walker, Director,
G. Abdallah, Lead Civil Engineer
S. Hassoun, Licensing Supervisor
R. Johnson, Nuclear Licensing Manager
B. Keck, Nuclear Engineering Manager
R. Salmon, Nuclear Compliance Supervisor

Nuclear Regulatory Commission

J. Giessner, Chief, Reactor Projects Branch 4

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000345/2011002-01	NCV	Failure to Fully Evaluate the Failure of H_2 O_2 Sampling Pump Trips During Calibration (1.R12.1)
05000341/2011002-02	NCV	Design Control Measures Failed to Ensure Adequacy of the Design Relating to the Reactor Building Crane Support Structure and Reactor Building Superstructure (1R20.1)
05000341/2011002-03	FIN	Reactor Scram due to Loss of Vacuum (4OA2.3)

Closed

05000341/2010011-01	URI	Reactor Building Crane Support Structure and Reactor Building Superstructure Did Not Meet Seismic Category I Requirements (1R20.1.b(1))
05000341/2007003-04	URI	Design Basis Changes for the Ultimate Heat Sink (4OA5.1)
05000341/2010-003-00	LER	Automatic Reactor Scram due to Degraded Condenser Vacuum (4OA3.2)

Discussed

None

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.

Section 1R01 – Adverse Weather Protection

- Memo, R Salmon to R Jones; Discussion on Snow Preparations; 02/04/2011

Section 1R04 – Equipment Alignment

- CARD 11-21461; NRC Concern: RHR Door D-12 Does not Latch Every Time When Pushed from CARDOX Room; 02/08/2011
- Drawing 6M721-5711-1; Reactor Water Clean-up Reactor Building; Revision AK
- Drawing 6M721-5721-2; Reactor Water Clean-up Filter Demineralizers; Revision T
- Drawing 6M721-5727; Reactor Building Closed Cooling Water; Revision W
- Drawing 6M721-5733-1; Fire Protection Functional Operating Sketch; Revision BB
- Drawing 6M721-5734; EDG System; Revision BB
- Procedure 23.501.01; Fire Water Suppression System; Revision 49
- Procedure 23.707, Attachment 2; RWCU System Electrical Lineup; 10/11/2006
- Procedure 23.707, Attachment 3; RWCU System Instrument Lineup; 06/14/2005
- Procedure 23.707, Attachment 1; RWCU System Valve Lineup; 01/20/2011
- Procedure 28.507.03; Fire Door Inspection; Revision 28
- Procedure 35.000.242; Barrier Identification/Classification; Revision 48

Section 1R05 – Fire Protection

- CARD 11-00081; Valve packing leak; 02/03/2011
- Drawing 6M721-5072, Fire Protection Plan Cable Tray Area, Revision K
- Fire Protection Pre Plan FP-RB-3-15a; Reactor Building Thermal Recombiner System Area, Zone 15, El. 641’6”; Revision 3
- Fire Protection Pre Plan FP-RB-4-17a; Reactor Building SLC System Zone 17 EL. 659’6”; Revision 3
- Fire Protection Engineering 09-0042; Acceptability of Fire Detector Spacing & Location Non-Conformance for Detection Zone 4; 02/19/2010
- Procedure FP-AB-2-8; Relay Room, Zone 8, El. 613’6”; Revision 6
- WO 29899663; Perform 28.508.01 Monthly Portable Fire Extinguisher Inspection; 12/02/2010
- WO 28149513; Perform 28.508.02 Fire Extinguisher Yearly Maintenance Inspection; 01/01/2010

Section 1R06 – Flood Protection

- UFSAR, Revision 16, dated October 2009, Section 2.4.2
- UFSAR, Revision 16, dated October 2009, Section 3.4.4
- UFSAR, Revision 16, dated October 2009, Section 3.1.2

Section 1R07 – Annual Heat Sink Performance

Section 1R11 – Licensed Operator Requalification Program

- Fermi 2 Evaluation Scenario SS-OP-904-1064; 64B Bus Trip, RR Pump B Seal Failure, HPCI Start Failure, LOCA; Revision 2, January 11, 2011

Section 1R12 – Maintenance Effectiveness

- 2010 Maintenance Rule Periodic Assessment TMIS 10-0093; 10/22/2010
- Briefing Sheet, CARD 11-23023; NRC Concern; T50N105B Flow Switch Calibrating Acceptance
- CARD 10-23218-01; Perform Peer Group Assessment of Issues Identified in Focused Self-assessment; 05/14/2010
- CARD 10-23219-03; Review Scope of systems; 11/20/2010
- CARD 10-29558; N20-K050, Cond Pmp Min Flow Controller Not Controlling at 4500 gpm; 10/26/2010
- CARD 10-30585; High Vibration on North Condenser Pump Motor during Uncoupled Run; 11/13/2010
- CARD 10-31311; Missed Opportunities on the North Condenser Motor Vibration Issues; 11/29/2010
- CARD 10-31430; Failure of CFD D Main Drain Caused Entry into TB Flooding AOP; 12/01/2010
- CARD 10-31443; N2000F610, Valve Failed to Full Open; 12/01/2010
- CARD 10-31732; High Vibration North Condenser Pump; 12/10/2010
- CARD 10-31910; Failure of the Calibration of NAHL-5/HS-L5S Flowmeter; 12/16/2010
- CARD 11-20215; Expert Panel Determined the N2000, Condensate System, A1; 01/07/2011
- CARD 11-21356; Received 4D3 (Unitized Actuator Bypass Valve Fault for the East Bypass Valve; 02/05/2011
- CARD 11-23023; NRC Concern, T50N105B Flow Stich Calibration Acceptance; 03/24/2011
- Enrico Fermi Maintenance Rule Periodic Assessment Report; September 2006 through August 2008
- MR Functional Failure Evaluation 101026-01, System ID N2000, Doc. ID 1345312; 11/03/2010
- MR Functional Failure Evaluation 101201-01, System ID N2000, Doc. ID 1347978; 12/16/2010
- MR Functional Failure Evaluation 101210-01, System ID N2000, Doc. ID 1348020; 12/22/2010
- MR Functional Failure Evaluation 101201-02, System ID N2000, Doc. ID 1347981; 12/16/2010

Section 1R13 – Maintenance Risk Assessments and Emergent Work Control

- Fermi 2 Plan of the Day; January 21, 24, 25, 27, 2011
- Fermi 2 Plan of the Day; February 28, 2011
- Fermi 2 Plan of the Day; March 1, 3, 4, 7, 8, 9, 10, 18, 21, 22, 23, 24, 2011
- Fermi 2 T+1 Performance Analysis Review; 02/28/2011

Section 1R15 – Operability Evaluations

- CARD 97-0908; Hydrogen Seal Oil Lip Ring End Metal Temperature High; 06/04/1997
- CARD 10-31357; Degraded motor bearing on South TBHVAC exhaust fan; 11/30/2010
- CARD 10-31496; NRC Concern with IRM Technical Specification Interpretation; 12/03/2010
- CARD 10-31614; 3D18 'IPCS Monitored Inputs Abnormal' alarms for Hydrogen Split Ring end TC92; 12/07/2010
- CARD 10-31926; Inappropriate Lube Requirements; 12/16/2010
- CARD 10-32197; Confirmation of Insulation Type Installed in Localized Recirculating Piping Areas within the Drywell; 12/28/2011
- CARD 11-20183; Procedure 35.000.217, Maintenance Lubrication, recommended Revision; 01/07/2011
- CARD 11-21136; Heat Shields found above Recirc Motor A; 02/01/2011
- CARD 11-21153; Errors in DC-5979 Vol 1; 0201/2011

- CARD 11-22407; T2300F450A Torus to Reactor Building Vacuum Breaker Did Not Indicate Properly during Surveillance Testing; 03/06/2011
- CARD 11-22500; 4D53 AVR General Alarm Received; 03/09/2011
- CARD 11-22541; 4D53-AVR General Alarm Received; 03/09/2011
- CARD 11-22827; Performance of 27.119 “Seal oil drain flow rate” Did Not Pass for the Gas Side / Exciter end; 03/18/2011
- Drawing 6I721-2678-03; Torus to Reactor Building Vacuum Breaker Valves T2300F450A thru T2300F410; Revision U
- Drawing 6M721-5739-1; Nitrogen Inerting System; Revision AA
- HSO oil temperatures, per IPCS/PI12/8/2010
- HSO TE (blue trace, N30DT2547) and SRE (pink trace, N30DT2548) T/Cs tracked closely until startup; 12/08/2010
- ODMI 10-010; High Hydrogen Seal Oil Temperature at MG SRE; 12/09/2010
- WO 30269249; Perform 24.402.01 Section 5.2 Reactor Building – Torus Vacuum Breaker Operability; 03/17/2011

Section 1R18 – Plant Modifications

- CARD 11-21658; Approved Scaffold Extension Does Not Have 10CFR50.59 Review Number; 02/13/2011
- CARD 11-21727; NRC Concern – Scaffolding in the Turbine Building; 02/15/2011
- CARD 11-21729; Permanent Scaffold Removal; 02/15/2011
- CARD 11-22248; NRC Concerns from reactor building walkdown; 02/28/2011
- NEI 96-07; Revision 1

Section 1R19 – Post-Maintenance Testing

- CARD 11-21484; Identify Impacts to the PM Technical Requirements or Critical Dates – RRMG B Scoop Tube; 02/09/2011
- CARD 11-21526; CSCCD-B31P404B Reactor Recirculating MG Set Scoop Tube Data Needs Updated; 02/09/2011
- CARD 11-21564; FR114 RRMG Sets ‘A’ and ‘B’ Work Performed by Potentially Unqualified Supplemental Craft; 02/10/2011
- CARD 11-22252; Leading Edge Flow Meter (LEFM) – Minor Alert Alarm Related to CPU Temperature; 02/28/2011
- CARD 11-21727; NRC Concern – Scaffolding in the Turbine Building; 02/15/2011
- EDP 36238; Feedwater Ultrasonic Flow Measurement System; Revision 0
- Infrequently Performed Test or Evolution 10-04; Post Installation Testing of EDP 36238 during the Operational Phase for Validating the Performance and Use of the New Leading Edge Flow Monitoring System; Revisions 1 and 2
- Procedure 24.109-02; Turbine Bypass Valve Operability Test; Revision 32
- Procedure 24.110.05; RPS-Turbine Control and Stop Valve Functional Test; Revision 42
- Procedure 46.000.200; Reactor Recirculation system MG Set Scoop Tube Positioner B3103B Calibration
- WO A533100100; Inspect/Test 260 VDC MCC Bucket MCC 2PB-1-10A; 02/25/2011
- WO A535100100; Inspect/Test 260 VDC MCC Bucket MCC 2PB-1-10B; 02/25/2011
- WO A539100100; Inspect/Test 260 VDC MCC Bucket MCC 2PB-1-12A; 02/25/2011
- WO 29636727; Perform 54.000.20 Reactor Recirculating System MG Set Scoop Tube Positioner Operability; 03/09/2011
- WO 29898593; Clean Air Side of Cooling Coil; 02/25/2011

- WO 30278671; Clean Drain Valve Cap, Disassemble, Inspect & Clean Condensate Drain Trap; 02/25/2011
- WO 30278577; Check Belts and Sample the Oil in the Compressor – Division 2 Control Air Compressor; 02/25/2011
- WO 30278609; Perform 24.202.08 Section 5.2 HPCI Pump LSFT and Operability Test at 1025 psig; 03/02/2011
- WO 30278677; Replace Suction and Discharge Valves with New or Refurbished Valves; 02/25/2011
- WO 30608821; EDP-36238 (Perform IPTE 10-04); 11/12/2009
- WO 30978098; Lubricate Blower Bearings, Inspect Belts, Check Bolts & Clean Unit; 02/26/2010
- WO 31052217; Replace Feedback Potentiometer in South RRMG Set Scoop Tube Positioner; 10/14/2010
- WO 31403313; Perform 24.110.05 RPS-TCV/TSV Channel Function; 03/23/2011
- WO 31611700; Verify Cooler T4100B030 Motor Mounting Bolt Thread Engagement; 02/25/2011
- WO 32140548; Shop Work – Inspect the Old Condenser Pump; 12/10/2010
- WO 32145027; Inspect N CNDS Pump First Stage Impeller; 12/13/2010
- WO 32333467; Perform Partial Surveillance 24.109.02 to Stroke East Bypass Valve; 02/08/2011
- WO 32358977; Replace Feedback Potentiometer in South RRMG Set Scoop Tube Positioner; 02/12/2011

Section 1R20 - Outage Activities

- Calculation No. CN-23934; Reactor Building Crane for Enrico Fermi Atomic Power Plant No. 2 Bridge and Trolley Structural Calculations; March 8, 1973
- Calculation No. DC-6019; Assessment of the Interior Columns for the Reactor Building Steel Superstructure Including Crane Lifted Load; Volume IA, Revision 0
- Calculation No. 4.02.01; Crane Girder Splice; Section 12, Revision 0
- Calculation No. 4.02.04; Superstructure Roof Framing Bracing System; Volume I, Revision A
- Calculation No. 4.02.09; Reactor Building Superstructure Steel Girt and Column Framing Design; Volume I, Revision A
- CARD 10-22393; NRC-ISFSI Issue Calculation Issues; 03/19/2010
- CARD 10-22958; NRC ISFSI Issue – Calculation 4.02.04; 04/06/2010
- CARD 10-22979; NRC ISFSI Issue-Inspector's Questions about Calculations DC-6019 and 4.02.04; 04/07/2010
- CARD 10-23882; NRC ISFSI Issue-RB5 Crane Calculation; 05/07/2010
- CARD 10-24166; NRC ISFSI Issue-CMTRs used in calculation 4.02.09; 05/19/2010
- CARD 10-26278; NRC ISFSI Issue-Stresses is DC-6019 and 4.02.09; 07/24/2010
- CARD 10-26691; ISFSI NRC Issue - Crane Runway Girder Splice; 08/04/2010
- CARD 10-28562; NRC Identified –Discrepancy in Calculation 4.02.04; 09/27/2010
- CARD 10-28649; ISFSI NRC Issue – Typo in Calc. DC-6465; 09/29/2010
- CARD 10-28978; NRC identified-addition error in calculation 4.02.04; 10/07/2010

Section 1R22 – Surveillance Testing

- Procedure 24.307.16; Emergency Diesel Generator 13 – Start and Load Test; Revision 49
- Procedure 24.307.17; Emergency Diesel Generator 14 – Start and Load Test; Revision 49
- Procedure 24.425.02; Containment Airlock Operability Test; Revision 23
- Procedure 44.010.109; IRM Functional Test – RPS Trip System B; Revision 44

- Procedure 44.030.001; ECCS – Core Spray System, Division 1, Logic Functional Test; Revision 37
- Procedure 44.030.056; ECCS – Reactor Recirculation Riser DP, Division 1 Functional Test; Revision 29
- System Health Fermi 2; Emergency Diesel Generators; PIS # R30
- WO 30365479; Perform 44.030.056 ECCS Reactor Recirculating Riser, DP Division 1, Functional Test; 03/21/2011
- WO 30973685; Perform 24.307.16, Section 5.1, EDG 13 Start and Load Test – Slow Start; 01/31/2011
- WO 31011611; Perform 24.307.17, Section 5.1, EDG 14 Start and Load Test – Slow Start;
- WO 31036218; Perform 44.030.001 ECCS – Core Spray System, Division 1, Logic Functional Test; 01/29/2011
- WO 31393720; Perform 44.010.109 IRM Functional, Trip system B with Rod Blocks (F.O.); 01/29/2011
- WO 32354416; Perform Control Rod Cycling Prior to S/U From Planned Outages; 02/09/2011
- WO 3328060528; Perform 24.425.02, Inner and Outer Containment Air Lock Operability Test (F.O.); 02/04/2011

Section 1EP6 - Drill Evaluation

- Fermi RERP Drill 2011, March 15, 2011

Section 4OA1 – Performance Indicator Verification

- CARD 11-22074; NRC identified issue: Error in ROP submitted generation data (CDE); 02/24/2011
- PI IE01; Unplanned Scrams per 7,000 Critical Hours, 03/25/2010, 06/06/2010, 10/24/2010
- PI IE03; Unplanned Power Changes per 7,000 Critical Hours, Operating Data Reports January through December 2010; Power Changes 03/05/2010, 03/22/2010, 03/23/2010, 06/18/2010, 08/28/2010, 09/18/2010, 09/25/2010, 12/07/2010
- PI IE04; Unplanned Scrams with Complications, Post-Scram Data and Evaluations; 03/25/2010, 06/06/2010, 10/24/2010

Section 4OA2 – Identification and Resolution of Problems

- CARD 08-27073; Revision to DC-0919 Results in Higher Division 2 Bus Voltage to Prevent Inadvertent Grid Trip; 10/24/2008
- CARD 10-20823; CDBI 2010 Concern Revise DC-0919 to Include Correct LTC Volts Per Tap; 01/29/2010
- CARD 10-21733; 2010 CDBI DC-0919 LTC and Motor Starting; 02/25/2010
- CARD 10-21792; 2010 CDBI EDP 35621 Backfit Mod Issue; 03/01/2010
- CARD 10-26632; Missed Opportunity Review for Component Design Basis Inspections Results; 08/03/2010
- CARD 11-22042; 120 KV Line Voltage Drop Monitoring; 02/23/2011
- DC-6447; Auxiliary Power System Analysis; Revision 0
- Drawing 61721-2215-02; Elementary Diagram Core Spray System Relay Logic Systems 1 and 2 and Relay Schedules; Revision AC
- Drawing 61721N-2572-18; Schematic Diagram 4160 ESS Diesel Bus 12EB Load Shedding Strings; Revision X
- Drawing 61721N-2572-20; Schematic Diagram 4160 V ESS Diesel Bus 14ED Load Shedding Strings; Revision Z

- CARD 11-24268, NRC Identified Observation CARD 10-26632 change from RCE
- Drawing 61721-2572-29; Schematic Diagram 4160V ESS Buses # 65E & 65F Load Shedding Strings; Revision N
- Drawing 61721-2578-05; Single Line Diagram Relaying and Metering 4160 Bus 64B; Revision S
- Drawing 61721-2578-11; Single Line Diagram Relaying and Metering Diagram 4160V ESS Bus 65F and 65T
- ECR-35621-1; Revision of Calc DC-0919 Vol I to Reflect Minimum and Maximum Error Evaluation and Motor Starting Transient Study; Revision A
- EDP 3521; Backfit Modification for Compliance with NRC BTP PSB-1 Position B.1.b.1; Revision A
- EDP-36014; Replacement of Existing Degraded Voltage Relays; Revision 0
- Drawing 6SD721-2500-01; One Line Diagram Plant 4160V & 480V System Service Unit 2; Revision AP
- Work Order 30101748; ITE Voltage Relay Testing for Bus 64B Relay ZN-27D; 10/6/2010
- Work Order 30101748; ITE Voltage Relay Testing for Bus 64B Relay YN-27D; 10/6/2010
- Work Order 30101748; ITE Voltage Relay Testing for Bus 64B Relay YZ-27B; 10/5/2010
- Work Order 30101748; ITE Voltage Relay Testing for Bus 64B Relay XY-27B; 10/5/2010

Section 4OA3 - Followup of Events and Notices of Enforcement Discretion

- CARD 11-21897; NRFP Tachometer ‘A’ erratic indication; 02/18/2011
- CARD 11-21893; North Reactor Feedpump speed oscillations result in manual tripping of pump; 02/17/2011

Section 4OA5 - Other Activities

- Procedure 23.203; RHR Complex Service Water Systems; Revision 101
- 29. EDM.14; Fire Header Management; Revision 4
- ARP 7D3; Division 1 RHR Reservoir Level Abnormal; Revision 15
- ARP 7D4; Division 2 RHR Reservoir Level Abnormal; Revision 16
- DC-5894, Volume 1; RHR Reservoir Replenishment Requirements; Revision 0
- SE 95-0017; UFSAR Change Request LCR 95-049-UFS; Revision 0

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
AISC	American Institute of Steel Construction
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CAP	Corrective Action Program
CARD	Corrective Action and Resolution Document
CCHVAC	Control Center Heating, Ventilation, and Air Conditioning
CDBI	Component Design Basis Inspection
CFR	Code of Federal Regulations
DRP	Division of Reactor Projects
ECCS	Emergency Core Cooling System
EDP	Engineering Design Package
EECW	Emergency Equipment Cooling Water
HPCI	High Pressure Coolant Injection
HPCV	High Pressure Turbine Control Valve
HPSV	High Pressure Turbine Stop Valve
HVAC	Heating, Ventilation, and Air Conditioning
ISFSI	Independent Spent Fuel Storage Installation
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
IRM	Intermediate Range Monitor
LER	Licensee Event Report
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NIAS	Non-interruptible Air System
NRC	U.S. Nuclear Regulatory Commission
PARS	Publicly Available Records System
PI	Performance Indicator
PI&R	Problem Identification and Resolution
PM	Preventative Maintenance
PMT	Post-Maintenance Testing
RBCCW	Reactor Building Closed Cooling Water
RCE	Root Cause Evaluation
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RPA	Rod Pattern Adjustment
RRMG	Reactor Recirculating Motor Generator
RWC	Reactor Water Cleanup
SDP	Significance Determination Process
SJAE	Steam Jet Air Ejector
SSO	Safety Sysstem Outage
SW	Service Water
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink
URI	Unresolved Item

WO	Work Order

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Sincerely,

/RA/

John B. Giessner, Chief
Branch 4
Division of Reactor Projects

Docket No. 50-341
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Letter to J. Davis from J. Giessner dated May 2, 2011.

SUBJECT: FERMI POWER PLANT, UNIT 2, INTEGRATED INSPECTION REPORT 05000341/2011002

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