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

October 31, 2012

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

SUBJECT: FERMI POWER PLANT, UNIT 2, INTEGRATED INSPECTION REPORT 05000341/2012004

Dear Mr. Plona:

On September 30, 2012, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Fermi Power Plant, Unit 2. The enclosed inspection report documents the inspection results which were discussed on October 11, 2012, with Mr. J. T. Conner, Site Vice-President, 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, one self-revealed and three NRC-identified findings of very low safety significance were identified. The findings involved violations of NRC requirements. Additionally, a licensee identified violation is listed in Section 4OA7 of this report. However, because of their very low safety significance, and because the issues were entered into your corrective action program, the NRC is treating the issues as non-cited violations (NCVs) in accordance with Section 2.3.2 of the NRC Enforcement Policy.

If you contest any of these non-cited violations, 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. 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 Agencywide Document Access and Management 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/***

Jamnes L. Cameron, Chief
Branch 6
Division of Reactor Projects

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

Enclosure: Inspection Report 05000341/2012004
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

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| Docket No: | 50-341 |
| License No: | NPF-43 |
| Report No: | 05000341/2012004 |
| Licensee: | Detroit Edison Company |
| Facility: | Fermi Power Plant, Unit 2 |
| Location: | Newport, MI |
| Dates: | July 1 through September 30, 2012 |
| Inspectors: | R. Morris, Senior Resident Inspector<br>R. Jones, Resident Inspector<br>J. Bozga, Reactor Engineer<br>T. Briley, Reactor Engineer<br>D. Kimble, Senior Resident Inspector, Davis-Besse<br>M. Phalen, Senior Health Physicist<br>J. Rutkowski, Project Engineer<br>P. Smagacz, Reactor Engineer |
| Observer: | S. Bell, Health Physicist |
| Approved by: | J. Cameron, Chief<br>Branch 6<br>Division of Reactor Projects |

Enclosure

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## SUMMARY OF FINDINGS

Inspection Report 05000341/2012004; 07/01/2012 – 09/30/2012; Fermi Power Plant, Unit 2; Outage Activities, Identification and Resolution of Problems.

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. One self-revealed and three NRC-identified findings of very low safety significance (Green) were identified. The findings 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.

**1.** NRC-Identified and Self-Revealed Findings

**Cornerstone: Initiating Events**

- Green. A finding of very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," was identified by the NRC inspectors. Specifically, the licensee failed to perform dimensional testing of the reactor pressure vessel head strongback and the steam dryer/steam separator lifting device required by American National Standards Institute (ANSI) N14.6-1978. In addition, the license failed to perform nondestructive testing of steam dryer/steam separator lifting device major load carrying welds and critical areas required by ANSI N14.6-1978. These issues were entered into the licensee's corrective action program.

  The finding was determined to be more than minor because the finding was associated with the Initiating Events Cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown. Specifically, the purpose of the dimensional testing of reactor pressure vessel head strongback and steam dryer/steam separator lifting device and nondestructive testing of the steam dryer/steam separator lifting device major load carrying welds and critical areas is to limit the likelihood of a reactor pressure vessel head strongback or steam dryer/steam separator lifting device structural component failure, and hence, to ensure safe load handling of heavy loads over the reactor core or over safety-related systems, structures and components. The inspectors determined the finding was of very low safety significance following a qualitative significance determination process review performed by the Region III Senior Risk Analyst. The inspector did not identify a cross-cutting aspect associated with this finding because the concern was related to licensing basis established in the 1980s, and thus was not necessarily indicative of current licensee performance. (Section 1R20)

- Green. A finding of very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified by the NRC inspectors for the failure to ensure the adequacy of the steam dryer/steam separator lifting device design. Specifically, the inspectors identified four examples where the licensee failed to perform adequate evaluations of the structural elements and structural connections in accordance with ANSI N14.6 requirements as defined in Updated Final Safety Analysis Report section 9.1.4.4. These issues were entered into the licensee's corrective action program.

The finding was determined to be more than minor because the finding was associated with the Initiating Events Cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown. Specifically, the purpose of the lifting device design requirements was to limit the likelihood of a structural component failure, and hence, to ensure safe load handling of heavy loads over the reactor core or over safety-related systems. The inspectors determined the finding was of very low safety significance following a qualitative significance determination process review performed by the Region III Senior Risk Analyst. The inspector did not identify a cross-cutting aspect associated with this finding because the concern was related to a calculation from the 1980s, and thus was not necessarily indicative of current licensee performance. (Section 1R20)

**Cornerstone: Mitigating Systems**

- Green. A finding of very low safety significance and an associated NCV of 10 CFR 50.55a(f), "Inservice testing requirements," and 10 CFR Part 50, Appendix B, Criteria V, "Instructions, Procedures, and Drawings," was identified by the NRC inspectors. Specifically, the licensee failed to perform a required comprehensive pump test for division 1 and 2 emergency equipment cooling water makeup pumps within 2 years of the start of the third inservice testing interval. The third inservice testing interval commenced on February 17, 2010, and included a requirement to perform a comprehensive pump test for the division 1 and 2 emergency equipment cooling water makeup pumps within two years and every two years thereafter. The required comprehensive pump tests were not performed prior to February 17, 2012.

  The finding was determined to be more than minor because the finding was associated with the configuration control attribute of the Mitigating Systems Cornerstone and impacted the cornerstone objective of ensuring the capability of systems to prevent undesirable consequences (i.e., core damage). This finding was determined to be of very low safety significance because, following IMC 0609, Appendix E, Table 4a, "Characterization Worksheet for Initiating Events, Mitigating Systems, and Barrier Integrity Cornerstones," all questions were answered "no." This finding has a cross-cutting aspect in the area of Human Performance, Decision Making, supervisory and management oversight aspect because the licensee failed to appropriately oversee the development and implementation of the comprehensive pump testing (H.4 (c)). (Section 4OA2)

- Green. A self-revealed finding of very low safety significance and an associated NCV of 10 CFR 50 Appendix B, Section V, "Instructions, Procedures, and Drawings," was identified for the failure to adequately prevent foreign material from entering the hydraulic control unit for control rod 10-35, which caused control rod 10-35 to fail to fully insert on October 24, 2010. Subsequently, on November 18, 2011, control rod 10-35 again failed to fully insert during scram time testing. The root cause team identified the presence of foreign organic material and concluded it had been present for a long time, i.e., at least since or prior to 2006, and this material was the cause of the deficient operation of control rod 10-35 in October 2010 and November 2011.

The inspectors determined this finding was more than minor because it was associated with the configuration control attribute of the Mitigating Systems Cornerstone and impacted the cornerstone objective of ensuring the capability of systems to prevent undesirable consequences (i.e., core damage). This finding was determined to be of very low safety significance because, following IMC 0609, Appendix E, Table 4a, "Characterization Worksheet for Initiating Events, Mitigating Systems, and Barrier Integrity Cornerstones," all questions were answered "no." There was no cross-cutting aspect for this finding and NCV because the foreign material entered hydraulic control unit 10-35 sometime prior to 2006; and, therefore, the foreign material exclusion program inadequacies do not represent current performance. (Section 4AO2)

**2.** Licensee-Identified Violations

One 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 shutdown, continuing forced outage 12-02 which commenced with the catastrophic failure of the south reactor feed pump turbine on June 25, 2012. The Unit remained shut down until July 22, 2012. The Unit increased power to 2 percent and returned to a shutdown condition to repair a valve motor. The Unit restarted on July 27, 2012, and achieved 68 percent power on July 30, 2012, using one main feedwater pump. The Unit maintained 68 percent power until September 14, 2012, when the Unit scrammed due to a fault in the 120 kV switchyard. The Unit commenced startup on September 19, 2012, achieved 68 percent power on September 20, 2012, and remained there for the remainder of the inspection period.

### REACTOR SAFETY

**Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity**

1R01 Adverse Weather Protection (71111.01)

.1 External Flooding

a. Inspection Scope

The inspectors evaluated the design, material condition, and procedures for coping with the design basis probable maximum flood. The evaluation included a review to check for deviations from the descriptions provided in the Updated Final Safety Analysis Report (UFSAR) for features intended to mitigate the potential for flooding from external factors. As part of this evaluation, the inspectors checked for obstructions that could prevent draining, checked that the roofs did not contain obvious loose items that could clog drains in the event of heavy precipitation, and determined that barriers required to mitigate the flood were in place and operable. Additionally, the inspectors performed a walkdown of the protected area to identify any modification to the site which would inhibit site drainage during a probable maximum precipitation event or allow water ingress past a barrier. The inspectors also walked down underground bunkers/manholes subject to flooding that contained multiple train or multiple function risk-significant cables. The inspectors also reviewed the abnormal operating procedure for mitigating the design basis flood to ensure it could be implemented as written.

This inspection constituted one external flooding 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:

- Mechanical isolation of main steam to the south reactor feed pump/turbine (RFPT) for Engineering Design Package (EDP) -36982;
- residual heal removal (RHR) system shutdown cooling during the forced outage; and
- high pressure coolant injection (HPCI).

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, 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 corrective action program 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.

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, first floor mezzanine, cable run area;
- auxiliary building, second floor, division 2 switchgear room;

- auxiliary building, third floor control room, including computer room; and
- turbine building, first floor, north/center/south heater drain pump rooms.

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 to this report, the inspectors verified fire hoses and extinguishers were in their designated locations and available for immediate use; fire detectors and sprinklers were unobstructed; transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified minor issues identified during the inspection were entered into the licensee's corrective action program. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

.2 Annual Fire Protection Drill Observation (71111.05A)

a. Inspection Scope

On September 26, 2012, the inspectors observed a fire brigade activation for a fire in the auxiliary building, east reactor protection system, division 1 motor generator set room, fire zone 14. Based on this observation, the inspectors evaluated the readiness of the plant fire brigade to fight fires. The inspectors verified the licensee staff identified deficiencies; openly discussed them in a self-critical manner at the drill debrief; and took appropriate corrective actions. Specific attributes evaluated were:

- proper wearing of turnout gear and self-contained breathing apparatus;
- proper use and layout of fire hoses;
- employment of appropriate fire fighting techniques;
- sufficient firefighting equipment brought to the scene;
- effectiveness of fire brigade leader communications, command, and control;
- search for victim and propagation of the fire into other plant areas; and
- drill objectives.

Documents reviewed are listed in the Attachment to this report.

These activities constituted one annual fire protection inspection sample 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. In addition, the inspectors reviewed licensee drawings to identify areas and equipment that may be affected by internal flooding caused by the failure or misalignment of nearby sources of water, such as the fire suppression or the circulating water systems. The inspectors also reviewed the licensee's corrective action documents with respect to past flood-related items identified in the corrective action program to verify the adequacy of the corrective actions. The inspectors performed a walkdown of the following plant areas to assess the adequacy of watertight doors and verify drains and sumps were clear of debris and were operable, and that the licensee complied with its commitments:

- turbine building basement;
- HPCI sub-basement; and
- auxiliary building first floor.

Specific documents reviewed during this inspection are listed in the Attachment to this report.

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 of Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

On July 11 and 18, 2012, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator just-in-time training for restart and for operations with the south reactor feed pump out of service to verify operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;

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

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

.2 Resident Inspector Quarterly Observation of Heightened Activity or Risk (71111.11Q)

a. Inspection Scope

On July 22 and 29, 2012, the inspectors observed activities in the main control room during entrance into Mode 2, performance of reactor startup and power ascension following forced outage 12-02, and rolling the main turbine and synchronizing the generator to the grid. These were activities that required heightened awareness or were related to increased risk. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of 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 performance in these areas was compared to pre-established operator action expectations, procedural compliance and task completion requirements. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly licensed operator heightened activity/risk 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:

- U4100 turbine building heating, ventilation, and air conditioning;
- main feedwater change with 1 pump operation; and
- E4100 HPCI.

The inspectors reviewed events such as where ineffective equipment maintenance had resulted or could result 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 corrective action program with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

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

.1 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

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

- risk during start of forced outage 12-02, and International Transmission Company deenergizing Swan Creek line;
- risk during startup and power ascension following forced outage 12-02 with contingency plan if Custer transformer 103 were lost and with emergency core cooling system (ECCS) reactor pressure vessel water level division calibration and functional test;
- risk during reactor core isolation cooling (RCIC) outage and Force-on-Force drill;
- risk during division RHR safety system outage with contingency if Custer transformer 103 were lost and with RHR pump D component outage; and
- risk during emergency equipment cooling water (EECW) piping replacement with combustion turbine generator 11-1 maintenance and a reactor scram.

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.

Specific documents reviewed during this inspection are listed in the Attachment to this report.

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

b. <u>Findings</u>

No findings were identified.

1R15 <u>Operability Determinations and Functional Assessments</u> (71111.15)

.1 <u>Operability Evaluations</u>

a. <u>Inspection Scope</u>

The inspectors reviewed the following issues:

- Operational Decision Making Issue (ODMI) 12-005, “Extended Plant Operation with Only the North RFPT;”
- corrective action and resolution document (CARD) 12-26703, “P4400F625B EECW Division 2 Make-up Pump Discharge Check Valve Failure;”
- CARD 12-26429, “In-Service Testing Program Comprehensive Pump Test for EECW Makeup Pump;” and
- CARD 12-27126, “Further Reduction in North RFP Moisture Level.”

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 TSs 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 four 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 modifications:

- EDP-36982, "South RFPT Mechanical Isolation;" EDP 36984, "South RFPT Electrical and Instrument and Controls (I&C) Isolations;" and 10 CFR 50.59 Evaluation for the modifications; and
- EDP 35633, "Integrated Plant Computer System (IPCS) Core Switch Replacement in Control Room Balcony."

The inspectors reviewed the configuration changes and associated 10 CFR 50.59 safety evaluation screening against the design basis, the UFSAR, and the TSs, as applicable, to verify the modification did not affect the operability or availability of the affected systems. 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 modifications with operations, engineering, and training personnel to ensure the individuals were aware of how the operation with the plant modifications 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 modification sample and one permanent 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 (PM) activities to verify procedures and test activities were adequate to ensure system operability and functional capability:

- WO 33798851, "Emergency Diesel Generator (EDG) 14 Air Intake Filter Clean and Replace;"
- WO 33931602, "EDP-35633 IPCS Core Switch Replacement in Control Room Balcony;"
- radiography for WO 34718804 and WO 34718834 for EDP-36982, "South RFPT Mechanical Isolations;"
- WO 34826050, "Replace Failed Motor in E-4150F002;"
- WO 34471516, "Troubleshooting/Repair Reactor Recirculation Motor Generator (RRMG) B Set Speed Oscillations PMT;"
- multiplexer (MUX) B failure and return to service;
- WO 34910165, "Install/remove monitor equipment for RRMG Set A;" and WO 35317095, "Inspect Breaker 65G-G3 for North RRMG Set Drive Motor."

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 the TSs, UFSAR, 10 CFR 50 requirements, licensee procedures, and various NRC generic communications to ensure the test results adequately ensured the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with PMTs to determine whether the licensee was identifying problems and entering them in the corrective action program 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 Forced Outage Activities

a. Inspection Scope

The inspectors evaluated outage activities for a forced outage that began on June 25, 2012, and continued through July 27, 2012. 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, personnel fatigue management, startup and heatup activities, and identification and resolution of problems associated with the outage. Also, the inspectors observed the isolation of the south RFP and testing for plant operations with one feedwater pump.

This inspection was counted as a sample in the previous quarter. Therefore, this does not constitute an outage sample for this quarter.

b. Findings

No findings were identified.

.2 Operating Experience Smart Sample 2007-003, Revision 2, "Crane and Heavy Lift Inspection, Supplemental Guidance for Inspection Procedure-71111.20"

a. Inspection Scope

From August 27 through September 14, 2012, the inspectors reviewed the licensee's Control of Heavy Loads Program in conjunction with the NRC's Operating Experience Smart Sample (OpESS) FY2007–03, Revision 2, "Crane and Heavy Lift Inspection, Supplemental Guidance for IP-71111.20," specifically related to the removal and installation of the reactor vessel head during refueling outages. The inspectors performed the activities listed below during the inspection. Documents reviewed during the inspection are listed in the Attachment of this report.

- Reviewed the licensee's reactor building crane preventative maintenance program procedures. Also, reviewed a sample of licensee records of reactor building crane inspections completed prior to reactor disassembly and reactor head lift.

- Reviewed the licensee's submittals and commitments related to Generic Letters 80–113 and 81–07, "Control of Heavy Loads."

- Reviewed the licensee's structural calculations for reactor building crane design to Seismic Category I requirements. Also, reviewed documents supporting the licensee's classification of the reactor building crane as single failure proof (refer to Inspection Report 05000341/2011002, Section 1R20).

- Reviewed a sample of the licensee's calculations of rigging and special lifting devices used to remove and install the reactor vessel head during refueling operations.

- Reviewed a sample of the licensee's procedures that control reactor vessel safe load path to remove and install the reactor vessel head during refueling operations.

- Reviewed the licensee's preventative maintenance, inspection, and testing program procedures of rigging and special lifting devices used to remove and install the reactor vessel head during refueling operations.

This inspection was done per NRC's OpESS FY2007-03, Revision 2, "Crane and Heavy Lift Inspection, Supplemental Guidance for IP-71111.20," and is a part of the outage inspection that was counted as a sample in the previous quarter.

b. Findings

1) Inspection Procedure for Reactor Pressure Vessel Head Strongback and Steam Dryer/Steam Separator Lifting Device Omitted Dimensional Testing and Nondestructive Testing Requirements

Introduction: The inspectors identified a finding of very low safety significance (Green) and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," in that, the procedure used to demonstrate compliance with ANSI N14.6-1978, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4500 kg) or More for Nuclear Materials," Section 5.3.1, provisions did not include dimensional testing of reactor pressure vessel (RPV) head strongback and steam dryer/steam separator lifting device. In addition, several major load carrying welds and critical areas of steam dryer/steam separator lifting device were not being nondestructively tested in accordance with the provisions specified in ANSI N14.6–1978. As a result, the licensee used the RPV head strongback and steam dryer/steam separator lifting device during each refueling outage without performing dimensional testing of RPV head strongback and steam dryer/steam separator lifting device and without performing nondestructive testing of major load carrying welds and critical areas of the steam dryer/steam separator lifting device.

Description: Section 3.2-1 of the UFSAR, classified both the RPV head strongback and steam dryer/steam separator lifting device as having quality assurance requirements in accordance with 10 CFR Part 50 Appendix B. Section 9.1.4.4 of the UFSAR states in part, "The special lifting devices at Fermi 2 are the reactor pressure vessel head strongback and the dryer/steam separator lifting device…. Periodic testing of these special lifting devices meets the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," by following ANSI N14.6-1978." Section 5.3.1 of ANSI N14.6–1978 requires each special lifting device be subjected to either a load test or dimensional testing, visual inspection, and nondestructive testing of major load carrying welds and critical areas. The licensee implemented the second option and hence did not perform a load test of the RPV head strongback and steam dryer/steam separator lifting device. Preventative Maintenance Activity No. F130, "Perform NDE on Special Lifting Device" identified the only major load carrying welds for the steam dryer/separator lifting device as part of detail 'A' of Fermi Drawing 6C721-2201, "Design and Details of Dryer/Separator Lifting Device," Revision B. The inspectors noted that Calculation

DC-3266, “Design Check for Rigging Apparatus,” Revision A, evaluated the design adequacy of the major load carrying welds on the steam dryer/steam separator lifting device. In addition to detail ‘A’, Calculation DC-3266 identified the major load carrying welds were also part of Section 1 and socket box detail of Drawing 6C721-2201. The licensee agreed Section 1 and socket box detail of Drawing 6C721-2201 were major load carrying welds as well. The inspectors also determined that Procedure 32.RIG.018, “Guidelines and Practices for the Use of Hoisting and Rigging Equipment,” Revision 8, did not include the ANSI N14.6–1978 requirement to perform dimensional testing of the RPV head strongback and steam dryer/steam separator lifting device. The licensee could not produce documentation to verify nondestructive testing of load carrying welds associated with Section 1 and socket box detail of Drawing 6C721-2201 welds were performed as well as dimensional testing of both the strongback and the lifting device. In response to these concerns, the licensee initiated CARD 12-27537, “NRC Identified NDE Inspections,” dated September 12, 2012, to address these concerns.

Analysis: The inspectors determined the failure to perform dimensional testing of RPV head strongback and steam dryer/steam separator lifting device and nondestructive testing of several of the steam dryer/steam separator lifting device major load carrying welds and critical areas was contrary to the ANSI N14.6-1978 provisions and was a performance deficiency. The finding was determined to be more than minor in accordance with IMC 0612, Appendix B, “Issue and Screening,” Minor Question 4, because the finding was associated with the Initiating Events Cornerstone attribute of equipment performance and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown. Specifically, the purpose of the dimensional testing and nondestructive testing of RPV head strongback and steam dryer/steam separator lifting device is to limit the likelihood of a structural component failure, hence, ensuring safe load handling of heavy loads over the reactor core or over safety-related systems.

The inspectors determined the finding could be evaluated using IMC 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” Exhibit 1, for the Initiating Events Cornerstone. The inspectors determined the finding would be related to a transient initiator, which could cause a reactor trip and loss of mitigation equipment. Therefore, the Region III senior reactor analyst (SRA) performed a more detailed risk evaluation. Since the plant would be shutdown when the RPV head strongback or steam dryer/steam separator lifting device was used, the RIII SRA conducted the assessment of the risk significance using IMC 0609, Appendix G, “Shutdown Operations SDP.” Table 1 of Appendix G, “Losses of Control,” does not apply since no event occurred. The SRA reviewed Appendix G, Attachment 1, “Shutdown Operations SDP, Phase I Operational Checklists for Both Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs).” The applicable checklist was Checklist 6, “BWR Cold Shutdown or Refueling Operation - Time to Boil < 2 hours: Reactor Coolant System level < 23' Above Top of Flange.” The applicable safety functions impacted were decay heat removal and inventory control. The SRA continued the risk-evaluation with a Phase II SDP assessment.

The SRA reviewed Appendix G, Attachment 3, “Phase II Significance Determination Process Template for BWR during Shutdown.” In Phase II, the most limiting plant operating state was POS-1 with an early time window. The SRA evaluated the impact of the performance deficiency on the loss of decay heat (i.e., loss of residual heat removal

(LORHR)) and loss of inventory initiators in Appendix G. Both initiators were representative of “condition” findings as defined in Appendix G, Attachment 3.

For each initiator, the estimated initiating event likelihood (IEL) was 3E-4/yr. The licensee provided data from Fermi's past two refueling outages that showed the RPV head strongback and steam dryer/steam separator lifting device are used to move a very heavy load a combined eight times per refuel outage (for approximately 10 hours per outage). At Fermi refuel outages are held every 18 months. Based on this, the number of lifts over a 12-month period was computed to be 5.3 lifts. The SRA also used information in NUREG 1774, “A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002,” to obtain an estimate of failure probability of the RPV head strongback and steam dryer/steam separator lifting device. Based on actual crane operating experience data from Commercial U.S. Nuclear Power Plants, this NUREG estimated the probability of load drops per demand for very heavy loads to be 5.6E-05. Thus the IEL of 3E-4/yr was estimated based on 5.3 lifts per year with a failure probability of 5.6E-5 per lift. The SRA noted the licensee visually inspected the strongback and lifting device prior to each use and performed limited nondestructive examination every five years with no deficiencies identified, so this IEL may be conservative for Fermi.

Regarding loss of inventory, the mitigating functions for this initiator were evaluated using Worksheet 1, “SDP for a BWR Plant - Loss of Inventory in POS 1.” The result of the loss of inventory initiator was 9E-9. In Worksheet 1, the dominant sequences involved failure of manual low and high pressure injection, and failure to isolate the loss of inventory or failure of reactor coolant system (RCS) pressure control.

Regarding loss of operating train of RHR, the mitigating functions for this initiator were evaluated using Worksheet 4, “SDP for a BWR Plant - Loss of Operating Train of RHR in POS 1.” The result of the LORHR initiator was 6E-10. In Worksheet 4, the dominant accident sequences involved failure of manual low and high pressure injection and failure of RHR recovery or failure of containment venting.

The total risk result of the internal event analysis is the sum of the individual results from the initiators above adjusted by the counting rule (i.e., multiply by 3.3) that is described in IMC 0609, Appendix A. The total internal event delta core damage frequency is on the order of 3.2E-8/yr. The SRA reviewed the licensee's risk evaluation. The licensee calculated a cumulative frequency of 8.4E-7/yr based on the load drop event tree in Figure 21 from NUREG 1774. This frequency was only the frequency that safe shutdown equipment is damaged from a load drop, not necessarily a core damage frequency.

Based on these results, the SRA determined the risk of this performance deficiency to be of very low safety significance (Green). The inspector did not identify a cross-cutting aspect associated with this finding because the concern was related to licensing requirements established from the 1980s, and thus was not necessarily indicative of current licensee performance.

<u>Enforcement</u>: Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and be accomplished in accordance with these instructions, procedures, or drawings. Section 5.3.1 of ANSI N14.6–1978 requires, “In cases where surface

cleanliness and conditions permit, the load testing may be omitted and dimensional testing, visual inspection, and nondestructive testing of major load-carrying welds and critical areas in accordance with Section 5.5 of this standard shall suffice."

Contrary to the above, the licensee failed to have activities prescribed in documented instructions or procedures to ensure the ANSI N14.6–1978 provisions to perform dimensional testing of the RPV head strongback and steam dryer/steam separator lifting device was performed. Also, the licensee failed to have activities prescribed in documented instructions or procedures to ensure the ANSI N14.6–1978 provisions to perform nondestructive testing of several specific load carrying welds of steam dryer/steam separator lifting device was performed. Specifically, these provisions were not included in Procedure 32.RIG.018 and Preventative Maintenance Activity F130. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as CARD 12–27537, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. Inspection Procedure for Reactor Pressure Vessel Head Strongback and Steam Dryer/Separator Lifting Device Omitted Testing Requirement (NCV 05000341/2012004-01).

2) Inadequate Evaluation of Steam Dryer/Steam Separator Lifting Device

Introduction: The inspectors identified a finding of very low safety significance (Green) and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for failure to provide adequate design control measures for the steam dryer/steam separator lifting device. Specifically, the inspectors identified four examples where the licensee failed to perform adequate evaluations of the lifting device in accordance with ANSI N14.6 provisions as defined in UFSAR Section 9.1.4.4. As a result, the design basis calculation was not sufficient to ensure conformance with UFSAR requirements for safe load handling of heavy loads over the reactor core or over safety-related systems.

Description: Section 3.2 of the UFSAR classified the steam dryer/steam separator lifting device as having quality assurance requirements in accordance with 10 CFR Appendix B. Section 9.1.4.4 of the UFSAR required the steam dryer/steam separator lifting device design to meet the provisions of ANSI N14.6-1978. Calculation DC-3266, "Design Check for Rigging Apparatus," Revision A, was used to demonstrate compliance with ANSI N14.6.

During a review of the aforementioned calculation the inspectors identified the following four representative examples in which the licensee failed to meet the ANSI N14.6 provisions:

1. ANSI N14.6-1978, Section 3.2.1.1, states in part, "The load-bearing members of a special lifting device shall be capable of lifting three times the combined weight of the shipping container with which it will be used, plus the weight of intervening components of the special lifting device." Calculation DC-3266 did not meet this provision in that the weight of the lifting device and its intervening components were not included in the design.

2. ANSI N14.6-1978, Section 3.2.1.1, states in part, "When materials that have yield strengths above 80 percent of their ultimate strength are used, each case requires special consideration, and the foregoing stress design factors do not apply. Design shall be on the basis of the material's fracture toughness, and the designer shall establish the criteria." The socket lifting pin and adapter lifting pin of the lifting device

had yield strengths greater than 80 percent of their ultimate strengths. Calculation DC-3266 did not design these structural elements based on their fracture toughness properties.

3. ANSI N14.6-1978, Section 3.2.1.1, states in part, "The load-bearing members of a special lifting device shall be capable of lifting three times the combined weight of the shipping container with which it will be used, plus the weight of intervening components of the special lifting device, without generating a combined shear stress or maximum tensile stress at any point in the device in excess of the corresponding minimum yield strength of their materials of construction." Calculation DC-3266 performed an analysis of the vertical stiffeners for the lifting device. The analysis showed the stiffeners have an applied stress greater than the allowable stress which would exceed the minimum yield strength of the stiffener material.

4. ANSI N14.6-1978, Section 3.2.1.1, states in part, "The load-bearing members of a special lifting device shall be capable of lifting three times the combined weight of the shipping container with which it will be used, plus the weight of intervening components of the special lifting device, without generating a combined shear stress or maximum tensile stress at any point in the device in excess of the corresponding minimum yield strength of their materials of construction. They shall also be capable of lifting five times that weight without exceeding the ultimate strength of the materials." Calculation DC-3266 did not analyze each and every structural element and connection for the stress design factors based on the yield and the ultimate strength of the material.

In response to this concern, the licensee initiated CARD 12-27276, "NRC Identified-Revise Calculation DC-3266, Volume I," dated August 31, 2012, to address these concerns.

Analysis: The inspectors determined the failure to meet the Section 3.2.1 design provisions of ANSI N14.6 for the steam dryer/steam separator lifting device was a performance deficiency.

The finding was determined to be more than minor in accordance with IMC 0612, Appendix B, "Issue and Screening," Minor Question 4, 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 plant stability and challenge critical safety functions during shutdown. Specifically, the purpose of the lifting device design requirements was to limit the likelihood of a structural component failure, and hence, to ensure safe load handling of heavy loads over the reactor core or over safety-related systems.

The inspectors determined the finding could be evaluated using IMC 0609, Appendix A, "The SDP for Findings At-Power," Exhibit 1, for the Initiating Events Cornerstone. The inspectors determined the finding would be related to a transient initiator which could cause a reactor trip and loss of mitigation equipment. Therefore, the Region III SRA performed a more detailed risk evaluation. The Region III SRA determined the risk significance documented in Section 1R20.2.b.1) of this report applies to this finding as well. As stated in Section 1R20.2.b.1), the SRA determined the risk of the finding to be of very low safety significance (Green). The inspector did not identify a cross-cutting aspect associated with this finding because the concern was related to a calculation from the 1980s, 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 calculation methods, or by the performance of a suitable testing program.

Contrary to the above, on August 31, 2012, the inspectors determined that for Calculation DC-3266 the licensee’s design control measures failed to ensure adequacy of the design. Specifically, this calculation did not conform to and was nonconservative with respect to ANSI N14.6 provisions.

Because this violation was of very low safety significance (Green) and it was entered into the licensee’s corrective action program as CARD 12-27276, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. Inadequate Evaluation of Steam/Dryer Steam Separator Lifting Device (NCV 05000341/2012004-02)

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 24.307.17, Section 5.1, “EDG 14 Start and Load Test – Slow Start” (routine);
- WO 30757211, “Perform 43.401.206 Section 6.1 and 6.2 Local Leak Rate Testing for Airlock (X-2)” (routine);
- Procedure 24.321.07, “Operability of 480V Swing Bus 72CF Automatic Throwover Scheme” (routine); and
- Procedure 24.202.08, “HPCI Time Response and Pump Operability Test at 1025 psi” (inservice testing (IST)).

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 IST activities, testing was performed in accordance with the applicable version of Section XI, American Society of Mechanical Engineers (ASME) code, and reference values were consistent with the system design basis;
- where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure;
- where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;
- prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;
- equipment was returned to a position or status required to support the performance of its safety functions; and
- all problems identified during the testing were appropriately documented and dispositioned in the corrective action program.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted three routine surveillance testing samples and one IST sample as defined in IP 71111.22-02 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 September 11, 2012, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the control room simulator and the 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 weaknesses 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 corrective action program. 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.

## 2. RADIATION SAFETY

**Cornerstone: Occupational Radiation Safety**

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

These inspection activities supplement those documented in Inspection Report 05000341/2012003 and constitute one complete sample as defined in IP 71124.01-05.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed all licensee performance indicators for the Occupational Exposure Cornerstone for follow-up. The inspectors reviewed the results of radiation protection program audits (e.g., licensee’s quality assurance audits or other independent audits). The inspectors reviewed any reports of operational occurrences related to occupational radiation safety since the last inspection. The inspectors reviewed the results of the audit and operational report reviews to gain insights into overall licensee performance.

b. Findings

No findings were identified.

.2 Radiological Hazard Assessment (02.02)

a. Inspection Scope

The inspectors determined if there have been changes to plant operations since the last inspection that may result in a significant new radiological hazard for onsite workers or members of the public. The inspectors evaluated whether the licensee assessed the potential impact of these changes and implemented periodic monitoring, as appropriate, to detect and quantify the radiological hazard.

b. Findings

No findings were identified.

.3 Instructions to Workers (02.03)

a. Inspection Scope

The inspectors reviewed selected occurrences where a worker’s electronic personal dosimeter noticeably malfunctioned or alarmed. The inspectors evaluated whether workers responded appropriately to the off-normal condition. The inspectors assessed whether the issue was included in the corrective action program and dose evaluations were conducted as appropriate.

b. Findings

No findings were identified.

.4 Contamination and Radioactive Material Control (02.04)

a. Inspection Scope

The inspectors reviewed the licensee's criteria for the survey and release of potentially contaminated material. The inspectors evaluated whether there was guidance on how to respond to an alarm that indicates the presence of licensed radioactive material.

The inspectors reviewed the licensee's procedures and records to verify the radiation detection instrumentation was used at its typical sensitivity level based on appropriate counting parameters. The inspectors assessed whether or not the licensee has established a de facto "release limit" by altering the instrument's typical sensitivity through such methods as raising the energy discriminator level or locating the instrument in a high-radiation background area.

The inspectors selected several sealed sources from the licensee's inventory records and assessed whether the sources were accounted for and verified to be intact.

The inspectors evaluated whether any transactions since the last inspection involving nationally tracked sources were reported in accordance with 10 CFR 20.2207.

b. Findings

No findings were identified.

.5 Radiological Hazards Control and Work Coverage (02.05)

a. Inspection Scope

The inspectors assessed whether radiation monitoring devices were placed on the individual's body consistent with licensee procedures. The inspectors assessed whether the dosimeter was placed in the location of highest expected dose or that the licensee properly employed an NRC-approved method of determining effective dose equivalent.

The inspectors reviewed the application of dosimetry to effectively monitor exposure to personnel in high-radiation work areas with significant dose rate gradients.

The inspectors examined the licensee's physical and programmatic controls for highly activated or contaminated materials (nonfuel) stored within spent fuel and other storage pools. The inspectors assessed whether appropriate controls (i.e., administrative and physical controls) were in place to preclude inadvertent removal of these materials from the pool.

b. Findings

No findings were identified.

.6 Risk-Significant High Radiation Area and Very High Radiation Area Controls (02.06)

a. Inspection Scope

The inspectors discussed with the radiation protection manager the controls and procedures for high-risk high radiation areas (HRAs) and very high radiation areas (VHRAs). The inspectors discussed methods employed by the licensee to provide stricter control of VHRA access as specified in 10 CFR 20.1602, "Control of Access to Very High Radiation Areas," and Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas of Nuclear Plants." The inspectors assessed whether any changes to licensee procedures substantially reduce the effectiveness and level of worker protection.

The inspectors discussed the controls in place for special areas that have the potential to become VHRAs during certain plant operations with first-line health physics supervisors (or equivalent positions having backshift health physics oversight authority). The inspectors assessed whether these plant operations require communication beforehand with the health physics group, so as to allow corresponding timely actions to properly post, control, and monitor the radiation hazards including re-access authorization.

The inspectors evaluated licensee controls for VHRAs and areas with the potential to become a VHRA to ensure an individual was not able to gain unauthorized access to the VHRA.

b. Findings

No findings were identified.

.7 Radiation Worker Performance (02.07)

a. Inspection Scope

The inspectors reviewed radiological problem reports since the last inspection that found the cause of the event to be human performance errors. The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. The inspectors discussed with the radiation protection manager any problems with the corrective actions planned or taken.

b. Findings

No findings were identified.

.8 Radiation Protection Technician Proficiency (02.08)

a. Inspection Scope

The inspectors reviewed radiological problem reports since the last inspection that found the cause of the event to be radiation protection technician error. The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems.

b. Findings

No findings were identified.

.9 Problem Identification and Resolution (02.09)

a. Inspection Scope

The inspectors evaluated whether problems associated with radiation monitoring and exposure control were being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee's corrective action program. The inspectors assessed the appropriateness of the corrective actions for a selected sample of problems documented by the licensee that involved radiation monitoring and exposure controls. The inspectors assessed the licensee's process for applying operating experience to their plant.

b. Findings

No findings were identified.

2RS2 Occupational As-Low-As-Is-Reasonably-Achievable Planning and Controls (71124.02)

These inspection activities supplement those documented in Inspection Report 05000341/2012003 and constitute a partial sample as defined in IP 71124.02-05.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed pertinent information regarding plant collective exposure history, current exposure trends, and ongoing or planned activities in order to assess current performance and exposure challenges. The inspectors reviewed the plant's three-year rolling average collective exposure.

The inspectors reviewed the site-specific trends in collective exposures (using NUREG-0713, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities," and plant historical data) and source term (average contact dose rate with reactor coolant piping) measurements (using Electric Power Research Institute TR-108737, "BWR Iron Control Monitoring Interim Report," issued December 1998, and/or plant historical data, when available).

The inspectors reviewed site-specific procedures associated with maintaining occupational exposures as low as is reasonably achievable (ALARA), which included a review of processes used to estimate and track exposures from specific work activities.

b. Findings

No findings were identified.

.2 Radiological Work Planning (02.02)

a. Inspection Scope

The inspectors selected the following work activities of the highest exposure significance:

- refuel floor activities including vessel assembly and disassembly;
- refuel floor activities including in-vessel inspections; and
- drywell access above 627' for the E2100F006A valve.

The inspectors reviewed the ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements. The inspectors determined whether the licensee reasonably grouped the radiological work into work activities based on historical precedence, industry norms, and/or special circumstances.

The inspectors assessed whether the licensee's planning identified appropriate dose mitigation features; considered alternate mitigation features; and defined reasonable dose goals. The inspectors evaluated whether the licensee's ALARA assessment had taken into account decreased worker efficiency from use of respiratory protective devices and/or heat stress mitigation equipment (e.g., ice vests). The inspectors determined whether the licensee's work planning considered the use of remote technologies (e.g., teledosimetry, remote visual monitoring, and robotics) as a means to reduce dose and the use of dose reduction insights from industry operating experience and plant-specific lessons learned. The inspectors assessed the integration of ALARA requirements into work procedure and radiation work permit documents.

The inspectors compared the results achieved (dose rate reductions, person-rem used) with the intended dose established in the licensee's ALARA planning for these work activities. The inspectors compared the person-hour estimates provided by maintenance planning and other groups to the radiation protection group with the actual work activity time requirements, and evaluated the accuracy of these time estimates. The inspectors assessed the reasons (e.g., failure to adequately plan the activity, failure to provide sufficient work controls) for any inconsistencies between intended and actual work activity doses.

The inspectors determined whether post-job reviews were conducted and if identified problems were entered into the licensee's corrective action program.

b. Findings

No findings were identified.

.3 Problem Identification and Resolution (02.06)

a. Inspection Scope

The inspectors evaluated whether problems associated with ALARA planning and controls were being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee's corrective action program.

b. Findings

No findings were identified.

## 4. OTHER ACTIVITIES

**Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Public Radiation Safety, and Occupational Radiation Safety**

4OA1 Performance Indicator Verification (71151)

.1 Mitigating Systems Performance Index - Emergency AC Power System (MS-06)

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index (MSPI) - Emergency AC Power System performance indicator for the period from the third quarter 2011 through the second quarter 2012. 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, MSPI derivation reports, issue reports, event reports, and NRC integrated inspection reports for the period of July 2011 through June 2012, to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. 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 MSPI emergency AC power system sample (MS-06) as defined in IP 71151-05.

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index - High Pressure Injection Systems (MS-07)

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI - High Pressure Injection Systems performance indicator for the period from the third quarter 2011 through the second quarter 2012. 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, MSPI derivation reports, event reports, and NRC integrated inspection reports for the period of July 2011 through June 2012 to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. 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 MSPI high pressure injection system sample (MS-07) as defined in IP 71151-05.

b. Findings

No findings were identified.

.3 Mitigating Systems Performance Index - Heat Removal System (MS-08)

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI - Heat Removal System performance indicator for the period from the third quarter 2011 through the second quarter 2012. 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, MSPI derivation reports, and NRC integrated inspection reports for the period of July 2011 through June 2012 to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. 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 MSPI heat removal system sample (MS-08) as defined in IP 71151-05.

b. Findings

No findings were identified.

.4 Reactor Coolant System Specific Activity (BI-01)

a. Inspection Scope

The inspectors sampled licensee submittals for the RCS specific activity PI for Fermi Power Plant for the period from the third quarter 2011 through the second quarter 2012. The inspectors used PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee's RCS chemistry samples, TS requirements, issue reports, event reports and NRC integrated inspection reports 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. In addition to record reviews, the inspectors observed a chemistry technician obtain and analyze an RCS sample. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one RCS specific activity sample as defined in IP 71151-05.

b. Findings

No findings were identified.

.5 Occupational Exposure Control Effectiveness (OR01)

a. Inspection Scope

The inspectors sampled licensee submittals for the Occupational Radiological Occurrences PI for the period from the third quarter 2011 through the second quarter 2012. The inspectors used PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee's assessment of the PI for occupational radiation safety to determine if indicator-related data was adequately assessed and reported. To assess the adequacy of the licensee's PI data collection and analyses, the inspectors discussed with radiation protection staff the scope and breadth of its data review and the results of those reviews. The inspectors independently reviewed electronic personal dosimetry dose rate and accumulated dose alarms and dose reports and the dose assignments for any intakes that occurred during the time period reviewed to determine if there were potentially unrecognized occurrences. The inspectors also conducted walkdowns of numerous locked high and VHRA entrances to determine the adequacy of the controls in place for these areas. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one occupational exposure control effectiveness sample as defined in IP 71151-05.

b. Findings

No findings were identified.

.6 Radiological Effluent TS/Offsite Dose Calculation Manual Radiological Effluent Occurrences (PR01)

a. Inspection Scope

The inspectors sampled licensee submittals for the Radiological Effluent TS/Offsite Dose Calculation Manual (ODCM) Radiological Effluent Occurrences Performance Indicator for the period from the third quarter 2011 through the second quarter 2012. The inspectors used PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee's issue report database and selected individual reports generated since this indicator was last reviewed to identify any potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose. The inspectors reviewed gaseous effluent summary data and the results of associated offsite dose calculations for selected dates to determine if indicator results were accurately reported. The inspectors also reviewed the licensee's methods for quantifying gaseous and liquid effluents and determining effluent dose. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one radiological effluent TS/ODCM radiological effluent occurrences sample as defined in IP 71151 05.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As part of the various baseline inspection procedures discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify they were being entered into the licensee's corrective action program at an appropriate threshold, adequate attention was being given to timely corrective actions, and adverse trends were identified and addressed. Attributes reviewed included: 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 corrective action program 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 follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. This review was accomplished through inspection of the station's daily condition report packages.

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

b. Findings

No findings were identified.

.3 Selected Issue Follow-Up Inspection: CARD 12-24565, Motor Generator Set Stops Incorrectly Set during Performance of Procedure 54.000.20

a. Inspection Scope

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

- CARD 12-24565, "MG Set Stops Incorrectly Set During Performance of 54.000.20."

The inspectors reviewed the CARD 12-24565 investigation, cause determination, corrective actions, the reactivity event classification for this event, and conducted interviews. Documents reviewed in this inspection are listed in the Attachment to this report.

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

b. Observations

Introduction: On May 16, 2012, procedure 54.000.20, "Reactor recirculation system MG (RRMG) set scoop tube positioner operability test," was performed to set mechanical, electrical, and distributed control system (DCS) stops for both the 'A' and the 'B' RRMG sets. Both RRMG set speeds and core flow were varied to collect data, establish a least squares fit of the data, and then extrapolate the data to establish the mechanical, electrical and DCS set points. On May 17, 2012, the residents reviewed a copy of the completed procedure and noted the extrapolated data points did not fall upon the least squares fit regression line. The resident identified this discrepancy to the reactor engineering supervisor.

Discussion: On May 16, 2012, procedure 54.000.20, "Reactor recirculation system MG set scoop tube positioner operability test," was performed to set mechanical, electrical, and DCS stops for both the 'A' and the 'B' RRMG sets. Both RRMG set speeds and core flow were varied to collect data. Procedure 54.000.20 requires the data collected in attachment 1 (i.e., by varying core flow and RRMG set speeds) to be plotted on a graph and extrapolated to determine the mechanical, electrical and DCS setpoints. Functionally equivalent computer-generated plots can be used and annotated. In this case, an Excel spreadsheet was used to input the data and establish the regression line. The Excel spreadsheet settings were such that the spreadsheet calculations for the extrapolated points were not updated in real-time. This resulted in printouts that were used to complete procedure 54.000.20 not having correctly propagated results, and therefore, the mechanical high speed stops, electrical stops, and DSC high speed limits were incorrectly set.

The engineers had never reviewed the overall plot of the data in completing procedure 54.000.20, but relied upon their use of an Excel spreadsheet. Consequently, the licensee identified this issue as a procedural violation.

This resulted in the 'A' RRMG set stops to have been set incorrectly and conservatively, while the 'B' RRMG set stops were set incorrectly and non-conservatively. The 'B' RRMG set stops was a near miss to Technical Requirements Manual specification 3.4.7.1 violation because the incorrect settings were below the value specified in Cycle 16, Core Operating Limits Report, Revision 0, Table 2.

Further, the residents reviewed the completed procedure 54.000.20 archived in the licensee's document management system. This procedure and work order had a designated review by reactor engineering required following the performance of the procedure. The residents were advised that the reactor engineering reviewer was knowledgeable of the issues raised in CARD 12-24565; however, there was no documentation provided of the deficiencies in setting the mechanical high speed stops, electrical stops, or DCS high speed limits, and no listing of CARD 12-24565 provided by the reviewer.

Conclusions: Incorrectly setting the RRMG set mechanical stops was a near miss to a Technical Requirements Manual violation. Additionally, failing to validate the data collected and extrapolated to establish the mechanical, electrical, and DCS stops was a fundamental engineering process weakness, which resulted in a human performance departmental reset. Finally, designated reviewers should critically identify deficiencies identified during the review process. This provides more complete documentation of issues and concerns, (i.e., including known deficiencies for archival purposes).

c. Findings

No findings were identified.

.4 Selected Issue Follow-Up Inspection: Failure to Perform American Society of Mechanical Engineers Inservice Testing Comprehensive Pump Tests Requirement for Emergency Equipment Cooling Water Make-up Pumps

a. Inspection Scope

The inspectors selected the investigation of the failure to perform the ASME IST required comprehensive pump tests (CPTs) for the division 1 and 2 EECW makeup pumps. The third IST interval commenced on February 17, 2010, and included a requirement to perform a CPT for the division 1 and 2 EECW makeup pumps within two years and every two years thereafter. The required CPT tests were not performed prior to February 17, 2012. The inspectors reviewed CARD 11-20585, which identified the CPT requirement; CARD 12-26429, which identified the difficulties in performance of the CPT and established a team to investigate the feasibility of performing the CPT for the EECW makeup pumps; CARD 12-26834, which documented the inability to perform the CPT for division 1 prior to August 15, 2012; and CARD 12-26860, which identified the NRC inspectors' concern with failure to perform the CPT in February.

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

b. Findings

Introduction: The inspectors identified a finding of very low safety significance (Green) and an associated NCV of 10 CFR 50.55a(f), “Inservice Testing Requirements,” and 10 CFR 50, Appendix B, Criteria V, “Instructions, Procedures, and Drawings,” for failure to perform a required CPT for division 1 and 2 EECW makeup pumps within two years of the start of the third IST interval. The third IST interval commenced on February 17, 2010, and included a requirement to perform a CPT for the division 1 and 2 EECW makeup pumps within two years and every two years thereafter. The required CPT tests were not performed prior to February 17, 2012.

Description: On February 17, 2010, Fermi 2 commenced their third IST interval. The IST program owner identified a requirement to perform a CPT for the division 1 and 2 EECW makeup pumps. The IST program owner reviewed the Operations and Maintenance (OM) Code and determined the CPT required testing the EECW makeup pumps at or near their best efficiency point. This required a significant operation or evolution (SOE) to perform the initial CPT to utilize as the baseline data, and the CPT would then be performed every two years thereafter. CARD 11-20585 was initiated to develop the testing plan and the procedure, identify and assemble the required test equipment, and conduct the SOE prior to February 17, 2012. During the timeframe CARD 11-20585 was initiated and approximately mid-February 2012 several meetings were conducted with the IST program owner, system engineering, plant support engineering, and operations with no conclusive actions agreed upon. In addition, neither of the two work orders for the performance of the CPT SOEs for the EECW makeup pumps were ever entered into the work control system with a schedule for the required date of February 17, 2012.

There is no evidence of the IST program owner evaluating TS 5.5.6, IST and inspection program, and surveillance requirement 3.0.2 and applying the specific provisions and limitations of surveillance requirement 3.0.2 to the required EECW makeup pump CPT SOEs. Likewise, there is no evidence of any documented evaluations of any IST program deficiencies or nonconformances regarding failure to perform the CPT SOEs on or prior to the due date.

Title 10 CFR 50.55a(f), “Inservice Testing Requirements,” provides requirements for IST. The licensee is currently in their third 120-month interval of their IST program, which commenced February 17, 2010. The code of record for the licensee is the ASME Operations and Maintenance of Nuclear Power Plants Code 2004 Edition with no addenda. The IST requirement for performance of a CPT for the division 1 and 2 EECW makeup pumps was not performed as required. No relief had been requested by the licensee. Technical Specification 5.5.6, “Inservice Testing and Inspection Program,” applied the provisions of surveillance requirement 3.0.2 to the testing frequencies listed. Again, however, there is no evidence that the IST program owner evaluated the specific provisions and limitations of surveillance requirement 3.0.2 and applied them to the required EECW makeup pump CPT SOEs.

In addition, 10 CFR 50 Appendix B, Section V, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Contrary to the above, the Engineering Support Conduct Manual, Chapter 23 (MES23), Section 3.8, “Inservice Inspection and Inservice Testing”, provides that, “…inservice testing shall be scheduled, conducted, and documented in accordance with the approved…IST plan. Contrary to the above, the required EECW makeup pump CPT SOEs were not scheduled or performed prior to February 17, 2012, and there was no documentation of this nonconformance.

Although not issued until after the February 16, 2012, due date of this surveillance, Enforcement Guidance Memorandum (EGM) 12-001, issued February 24, 2012, provides enforcement discretion to TS Section 5.0, since the revision to the standard TSs to make the provisions of SR 3.0.2 (and SR 3.0.3) applicable also made the Section 3.0, “Surveillance requirement applicability,” applicable to the TS Section 5.0. The result was that applying the standard TS rules of usage would prohibit licensees from using the SR 3.0.2 (and 3.0.3) allowances for TS Section 5.0. The EGM further points out the OM Code does not make available test allowances similar to either SR 3.0.2 (or SR 3.0.3) under 10 CFR 50.55a(f). Note, the OM Code committee has recently developed a code case which models SR 3.0.2 to make available up to an additional 25 percent of the test frequency. As yet, however, the code case has neither been approved and issued by the ASME nor endorsed by the NRC. In this situation there is no evidence the IST program owner evaluated the provisions and limitations of SR 3.0.2 and applied them to the required EECW makeup pump CPT SOEs. Therefore, the provisions of EGM 12-001 do not appear to apply.

Analysis: The inspectors determined the failure to perform the ASME OM Code IST required CPT for the division 1 and 2 EECW makeup pumps was a performance deficiency that required evaluation using the SDP. The inspectors determined this finding was more than minor because it was associated with the configuration control attribute of the Mitigating Systems Cornerstone and impacted the cornerstone objective of ensuring the capability of systems to prevent undesirable consequences (i.e., core damage). This finding was determined to be of very low safety significance because, following IMC 0609, Appendix E, Table 4a, “Characterization Worksheet for Initiating Events, Mitigating Systems, and Barrier Integrity Cornerstones,” all questions were answered no. Therefore, the finding was determined to be of very low safety significance (Green).

This finding has a cross-cutting aspect in the area of Human Performance, Decision Making, supervisory and management oversight aspect because the licensee failed to appropriately oversee the development and implementation of the CPT (H.4 (c)).

Enforcement: Title 10 CFR 50.55a(f), “Inservice Testing Requirements,” provides requirements for IST. The licensee is currently in their third 120-month interval of their IST program, which commenced February 17, 2010. The code of record for the licensee is the ASME OM Code 2004 Edition with no addenda. The IST requirement for performance of a CPT for the division 1 and 2 EECW makeup pumps was not performed as required.

Further, 10 CFR 50 Appendix B, Section V, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawing of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. The Engineering Support Conduct Manual, Chapter 23 (MES23), Section 3.8, “Inservice Inspection and Inservice Testing,” provides that, “…inservice testing shall be scheduled, conducted, and documented in accordance with the approved…IST Plan." Contrary to the above, required EECW makeup pump CPTs were not scheduled or performed prior to February 17, 2012, and there was no documentation of this nonconformance.

Because the violation was of very low safety significance and it was entered into your corrective action program (CARD 12-26429 and CARD 12-26860), this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. Failure to Perform ASME In-service Testing Comprehensive Pump Testing (NCV 05000341/2012004-03).

.5 Selected Issue Follow-up Inspection: Failure of Control Rod 10-35 to Fully Scram during Scram Time Testing

(Closed) URI 05000341/2012003-01; Control Rod 10-35 Failure to Scram

a. Inspection Scope

The inspectors selected the investigation of the failure of control rod 10-35 to fully scram during scram time testing conducted on November 18, 2011. CARD 11-30357 was issued and the investigation identified foreign organic material in the inlet of the scram outlet valve, C11-F127. Previously, control rod 10-35 had failed to fully insert following an automatic reactor scram on October 24, 2010 (CARD 10-29509). The apparent cause evaluation attributed this failure to a hydraulic lock caused by blockage in the flow path between the control rod drive mechanism and the scram discharge volume. The inspectors reviewed the CARD 11-30357 root cause evaluation, and the CARD 10-29509 apparent cause evaluation. They directly observed various portions of the investigation to identify the deficiencies in the performance of control rod 10-35. Documents reviewed in this inspection are listed in the Attachment.

This inspection is a continuation of a selected issue followup inspection initiated and counted as a sample in the first quarter (Inspection Report 05000341/2012002).

b. Findings.

Introduction: A self-revealed finding of very low safety significance and an associated NCV of 10 CFR 50 Appendix B, Section V, "Instructions, Procedures, and Drawings," was identified for the failure to adequately prevent foreign material from entering the hydraulic control unit for control rod 10-35, which caused control rod 10-35 to fail to fully insert on October 24, 2010. Subsequently, on November 18, 2011, control rod 10-35 again failed to fully insert during scram time testing. The root cause team identified the presence of foreign organic material and concluded it had been present for a long time (i.e., at least since or prior to 2006), and this material was the cause of the deficient operation of control rod 10-35 in October 2010 and November 2011.

Description: On October 24, 2010, control rod 10-35 failed to insert upon actuation of an automatic reactor scram caused by loss of condenser vacuum (CARD 10-29509). An emergent issue team was formed to investigate this event. The apparent cause was determined to be a hydraulic lock caused by blockage in the flow path between the control rod drive mechanism and the scram discharge volume. The licensee never found any foreign material, but postulated that the likely foreign material was discharged into the scram discharge volume, ultimately ending up in the torus room sump. As a corrective action, for cycle 15 the licensee increased the frequency of performing TS surveillance 3.1.4.2 scram time testing to every 100 days, adjusted the representative sample size to assure all rods would be tested during cycle 15, and included control rod 10-35 in each quarterly scram time testing sample.

On November 18, 2011, during scram time testing, control rod 10-35 again failed to fully insert. CARD 11-30357 was issued and the root cause evaluation team identified the direct cause of the failure as foreign organic material found in the inlet of the scram outlet valve, C11-F127. The root cause was identified as less than adequate barriers for preventing the entry of foreign material into the control rod drive hydraulic system. The root cause evaluation team further determined there had been significant program changes to the foreign material exclusion program since 2006, which were sufficient to prevent foreign material from entering hydraulic control unit 10-35.

Further, the inspectors found that Quality Assurance Conduct Manual, Chapter 11 (MQA11), Enclosure B, Section 3.0, "Significant Condition Adverse to Quality Determination Criteria," provides the specific criteria for significant condition adverse to quality determination. This determination for CARD 11-30357 had been made by the CARD review board prior to completion of the root cause evaluation. The inspectors observed two criteria from MQA11, Enclosure B, Section 3.0 which seemed to fit the findings regarding the investigation into the failure of control rod 10-35 to fully insert; MQA11, Enclosure B, Section 3.0, item 6., repetitive items, or MQA11, Enclosure B, Section 3.0, item 3, deviations that require extensive evaluation. The investigation details appeared to meet both of these criteria.

Analysis: The inspectors determined that failure to prevent the entry of foreign material into the hydraulic control unit of control rod 10-35 was a performance deficiency that required evaluation using the SDP. The inspectors determined this finding was more than minor because it was associated with the configuration control attribute of the Mitigating Systems cornerstone and impacted the cornerstone objective of ensuring the capability of systems to prevent undesirable consequences (i.e., core damage). This finding was determined to be of very low safety significance because, following IMC 0609, Appendix E, Table 4a, "Characterization Worksheet for Initiating Events, Mitigating Systems, and Barrier Integrity Cornerstones," all questions were answered "no." Therefore, the finding was determined to be of very low safety significance (Green).

There is no cross-cutting aspect for this finding and NCV because the foreign material entered hydraulic control unit 10-35 sometime prior to 2006; and therefore, the foreign material exclusion program inadequacies do not represent current performance.

Enforcement: Title 10 CFR 50 Appendix B, Section V, "Instructions, Procedures, and Drawings," requires, in part, "activities affecting quality shall be prescribed by documented instructions, procedures, or drawing of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Contrary to the above, the root cause evaluation identified less-than-adequate foreign material exclusion program requirements (that had been improved subsequent to 2006) and failed to prevent the entry of foreign material into the hydraulic control unit for control rod 10-35.

Because the violation was of very low safety significance and it was entered into your corrective action program as CARD 11-30357, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. Failure of Control Rod 10-35 to Fully Scram during Scram Time Testing (NCV 05000341/2012004-04).

This finding and NCV closes unresolved item (URI) 05000341/2012003-01, “Control Rod 10-35 Failure to Scram.”

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

.1 (Closed) Licensee Event Report 05000341/2012-001; Loss of Shutdown Cooling Due to a Voltage Transient

On April 11, 2012, Operations was clearing tags in preparation for bus 65E restoration. Step 61 of STR 2012-001122 had connected a ground truck in bus 65E position E4 and installed a red danger tag. The ground truck was installed to perform work on bus 65E and load shedding string. The ground truck was not removed prior to energizing the bus. The breaker installed in bus 65E position E9 immediately tripped due to a fault and as a result of the electrical transient, the ‘A’ RHR pump tripped while operating in shutdown cooling mode with the plant in Mode 5 for RFO-15. The pump trip was caused by an isolation of the division 1 RHR shutdown cooling inboard isolation valve E1150F009. A self-revealed Green finding and associated NCV of 10 CFR 50 Appendix B, Section V, “Instructions, Procedures, and Drawings,” was identified for this event and included in Inspection Report 05000341/2012003, section 4OA3. No additional findings were identified following review of this licensee event report (LER). Documents reviewed as part of this inspection are listed in the Attachment to this report. This LER is closed.

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

.2 (Closed) Licensee Event Report 05000341/2012-002; Reactor Scram during Reactor Pressure Vessel Hydrostatic Test

On April 26, 2012, surveillance procedure 24.137.21, “Reactor Pressure Vessel System Leakage Test” was in progress with the plant operating in Mode 4 (Cold Shutdown). All control rods were inserted. The surveillance was classified as an infrequently performed test or evolution, because of the safety significance of the testing and since the test is performed only once per cycle. Control rod scram time testing and excess flow check valve testing were also in progress, as allowed by the precautions and limitations of the RPV system leakage test procedure.

A dedicated pressure control operator was assigned to maintain RPV pressure in the band of 1030 to 1055 psig as specified by the RPV system leakage test procedure. Pressure was controlled by using a combination of reactor water cleanup blowdown and control rod drive system flow adjustments. The dedicated pressure control operator selected a process computer point to display on a highly visible screen near the pressure control station. The selected point, B21CP6601, was displaying a reactor pressure average using two inputs. Due to ongoing excess flow check valve testing, one input to the average reactor pressure computer point became invalid as its instruments were removed from service prescribed by the excess flow check valve test. As a result, RPV pressure as seen by the process computer point, B21CP6601, lowered as isolated instruments slowly bled down (i.e., relieved pressure). Based on this false indication, the dedicated pressure control operator informed the control room supervisor that RPV pressure was lowering. At this time, actual reactor pressure was slowly rising. No adjustments were made based on these indications; and after approximately three minutes, RPV pressure reached the high pressure scram setpoint (1093 psig) causing a reactor scram. A self-revealed Green finding and associated NCV of TS 5.4.1.a for the licensee's failure to establish and implement procedures recommended by Regulatory Guide 1.33, was identified for this event and included in Inspection Report 05000341/2012003, section 4OA3. No additional findings were identified following review of this LER. Documents reviewed as part of this inspection are listed in the Attachment to this report. This LER is closed.

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

.3 (Discussed) Licensee Event Report (LER) 05000341/2012-003; Reactor Scram Due to Degrading Condenser Vacuum

This event occurred on June 25, 2012. After completing repairs to main unit transformer 2B, reactor power was raised to approximately 22 percent and the unit was synchronized to the power grid. Shortly after operations began to increase power, multiple vibration-related alarms were received for the south reactor feed pump, and the pump tripped. The south reactor feed pump had catastrophically failed and as a result, condenser vacuum was decreasing. Operations performed a manual scram by taking the mode selector switch to shutdown. All automatic actuations and isolations occurred as designed.

The unit remained shut down in forced outage 12-02 and plant configuration changes were installed to isolate the south reactor feed pump from plant systems. The unit was restarted on July 22, 2012, increased power to 2 percent, and returned to a shutdown condition to repair a valve motor. The unit subsequently restarted on July 27, 2012, and achieved 68 percent power on July 30, 2012, using the north reactor feed pump. The licensee has not completed their root cause investigation of this event. The review of this LER cannot be completed until the root-cause evaluation is completed and the corrective actions are reviewed. Documents reviewed as part of this inspection are listed in the Attachment to this report. This LER remains open.

This event follow-up review did not constitute a sample as defined in IP 71153-05

.4 Automatic Reactor Scram Due to the Loss of the 120 kV Switchyard

a. Inspection Scope

The inspectors reviewed the plant's response to an event occurring on September 14, 2012, resulting in an automatic reactor scram due to the loss of the onsite 120 kV switchyard. The loss of the 120 kV switchyard resulted in a loss of feedwater and condensate systems. The EDGs automatically started and loaded to provide electrical power to safety systems. Reactor water level decreased to 98 inches, at which point HPCI and RCIC initiated, and reactor water level was recovered. All plant systems responded accordingly with the scram. Offsite power was restored to electrical buses in a few hours. Documents reviewed in this inspection are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

4OA5 Other Activities

.1 (Discussed) NRC Temporary Instruction (TI) 2515/187, Inspection of Near-Term Task Force Recommendation 2.3 Flooding Walkdowns

a. Inspection Scope

Inspectors accompanied the licensee on a sampling basis during their flooding walkdowns, to verify the licensee's walkdown activities were conducted using the methodology endorsed by the NRC. These walkdowns are being performed at all sites in response to a letter from the NRC to licensees entitled, "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," dated March 12, 2012 (ADAMS Accession No. ML12053A340).

Enclosure 4 of the letter requested licensees to perform external flooding walkdowns using an NRC-endorsed walkdown methodology (ADAMS Accession No. ML12056A050). Nuclear Energy Industry document 12-07 titled, "Guidelines for Performing Verification Walkdowns of Plant Protection Features," (ADAMS Accession No. ML12173A215) provided the NRC-endorsed methodology for assessing external flood protection and mitigation capabilities to verify plant features credited in the current licensing basis for protection and mitigation from external flood events, are available, functional, and properly maintained.

b. Findings

Findings or violations associated with the flooding walkdowns, if any, will be documented in the fourth quarter integrated inspection report.

.2 (Discussed) NRC Temporary Instruction (TI) 2515/188, Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns

a. Inspection Scope

Inspectors accompanied the licensee on a sampling basis during their seismic walkdowns, to verify the licensee's walkdown activities were conducted using the methodology endorsed by the NRC. These walkdowns are being performed at all sites in response to a letter from the NRC to licensees, entitled "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," dated March 12, 2012 (ADAMS Accession No. ML12053A340).

Enclosure 3 of the March 12, 2012, letter requested licensees to perform seismic walkdowns using an NRC-endorsed walkdown methodology. Electric Power Research Institute document 1025286 titled, "Seismic Walkdown Guidance," (ADAMS Accession No. ML12188A031) provided the NRC-endorsed methodology for performing seismic walkdowns to verify plant features, credited in the current licensing basis for seismic events, are available, functional, and properly maintained.

b. Findings

Findings or violations associated with the seismic walkdowns, if any, will be documented in the fourth quarter integrated inspection report.

4OA6 Management Meetings

.1 Exit Meeting Summary

On October 11, 2012, the inspectors presented the inspection results to Mr. J. T. Conner, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

Interim exits were conducted for:

- The inspection results for the areas of Radiological Hazard Assessment and Exposure Controls; Occupational ALARA Planning and Controls; and RCS Specific Activity, Occupational Exposure Control Effectiveness, and RETS/ODCM Radiological Effluent Occurrences PI verification with Mr. J. T. Conner, Site Vice President, on September 14, 2012; and
- The inspection results for the "Crane and heavy lift inspection" with Mr. J. T. Conner, Site Vice President, on September 14, 2012.

The inspectors confirmed that none of the potential report input discussed for the Radiological Hazard Assessment and Exposure Controls; Occupational ALARA Planning and Controls; and RCS Specific Activity, Occupational Exposure Control Effectiveness, and RETS/ODCM Radiological Effluent Occurrences PI verification was considered proprietary. The inspectors confirmed that for the OpESS FY2007-003, "Crane and heavy lift inspection," all paper copies of proprietary documents would be shredded and all electronic files of proprietary documents would be deleted.

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 the NRC Enforcement Policy for being dispositioned as an NCV.

- Fermi TS 5.7.1(b) for high radiation area (HRA) controls states, in part, "Entry into such areas…may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them." Contrary to the above, on April 6, 2012, two individuals entered an HRA in the torus room without being briefed on their radiological conditions. This issue was documented in the licensee's corrective action program in CARD 12-22833. Immediate corrective actions included briefing the workers involved on their radiological conditions and verifying their individual accumulated radiological exposure. The finding was determined to be of very low safety significance because it was not an ALARA planning issue, there was no overexposure nor potential for overexposure, and the licensee's ability to assess dose was not compromised.

ATTACHMENT: SUPPLEMENTAL INFORMATION

## SUPPLEMENTAL INFORMATION

## KEY POINTS OF CONTACT

Licensee
T. Connor, Site Vice President
K. Scott, Plant Manager
Z. Rad, Nuclear Licensing Manager
B. Keck, Nuclear Engineering Manager
R. Laburn, Radiation Protection Manager
R. Salmon, Nuclear Compliance Supervisor
P. J. Pendergast, Principal Engineer, Licensing

Nuclear Regulatory Commission
Jamnes L. Cameron, Chief, Reactor Projects Branch 6

## LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

| | | |
|---|---|---|
| 05000341/2012004-01 | NCV | Inspection Procedure for Reactor Pressure Vessel Head Strongback and Steam Dryer/Separator Lifting Device Omitted Testing Requirements |
| 05000341/2012004-02 | NCV | Inadequate Evaluation of Steam Dryer/Steam Separator Lifting Device |
| 05000341/2012004-03 | NCV | Failure to Perform ASME Inservice Testing Comprehensive Pump Test Requirement |
| 05000341/2012004-04 | NCV | Failure of Control Rod 10-35 to Fully Scram during Scram Time Testing |

Closed

| | | |
|---|---|---|
| 05000341/2012003-01 | URI | Control Rod 10-35 Failure to Scram |
| 05000341/2012-001 | LER | Loss of Shutdown Cooling due to Voltage Transient |
| 05000341/2012-002 | LER | Reactor Scram during Reactor Pressure Vessel Hydrostatic Test |

Discussed

| | | |
|---|---|---|
| TI 2515/187 | TI | Inspection of Near-Term Task Force Recommendation 2.3 Flooding Walkdowns |
| TI 2515/188 | TI | Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns |
| 05000341/2012-003 | LER | Reactor scram due to degrading condenser vacuum |

## LIST OF DOCUMENTS REVIEWED

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

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

- Drawing 6A721AA-2001 – Hoyem-Basso Associates, Inc.; Partial Site Plan and Details; Revision A
- Fermi 2 UFSAR 2.4.2.2.1; Conditions Considered; Revision 16
- Fermi 2 UFSAR 2.4.5.6.2; Maximum Run-up Elevations; Revision 16
- NRC Contact Form, B Keck to M Morris; Elevation of EOF Relative to Its Design Basis Flood Height from Flooding; 08/16/2012

<u>1R04 – Equipment Alignment</u>

- Drawing 6M721-5708-1; High Pressure Coolant Injection System; Revision AN
- Procedure 23.205; Residual Heat Removal System; Revision 120

<u>1R05 – Fire Protection</u>

- Drawing 6A721-2407; Fire Protection Evaluation Reactor and Auxiliary Buildings Third Floor Plan, El. 641'-6" and 643'-6"; Revision S
- Drawing 6A721-2408; Fire Protection Evaluation Reactor and Auxiliary Buildings Fourth Floor Plan, El. 659-6"; Revision U
- Procedure 20.000.22; Plant Fired; Revision 42
- Procedure 28.508.01; Fire Protection Procedure, Monthly Portable Fire Extinguisher Inspection; Revision 20
- Procedure 28.508.02; Fire Protection Procedure, Fire Extinguisher Yearly Maintenance Inspection; Revision 20
- Procedure FP-AB-3-14c; Fire Protection Pre Plan, Auxiliary Building, East Reactor System Division 1 Motor Generator Set Room, Zone 14, El 643'6", Division II Switchgear Room, Zone 14, El 643'6"; Revision 3
- Procedure FP-AB-3M-13; Fire Protection Pre Plan, Computer Room, Zone 13, El. 655'6"; Revision 5
- Procedure FP-TB; Fire Protection Pre-Plan, Turbine Building; Revision 8

<u>1R06 – Flood Protection</u>

- Procedure ARP 2D76; Reactor Building NE Leakage to Floor Drain Sump High; Revision 6
- Procedure ARP 2D78; Reactor Building Floor/Equip Drain Sumps Level Hi-Hi-Lo-Lo; Revision 15
- Procedure ARP 2D105; Reactor Building Corner Rooms/HPCI Room Flood Level; Revision 13
- Procedure 20.000.01; Acts of Nature; Revision 44
- Procedure 20.000.03; Turbine Building Flooding; Revision 11

1R11 – Licensed Operator Requalification Program

- Nuclear Training Lesson Plan LP-GN-909-1121N; Startup Training / SRFP OOS; Revision 0
- Procedure 22.000.02; Plant Startup to 25 Percent Power; Revision 84
- Procedure 23.138.01; Reactor Recirculation System; Revision 106
- Startup Checklist; Cycle 16 Forced Outage 12-02 Startup; 07/13/2012

1R12 – Maintenance Effectiveness

- Apparent Cause Evaluation, CARDs 10-31618 and 11-21893; North Reactor Feedwater Pump Oscillations; 05/03/2012
- CARD 10-22001; Failure of Center TBHVAC Exhaust Fan; 03/07/2010
- CARD 11-20151; Expert Panel Determined the U4100 System A1; 01/06/2011
- EDP-36673; Replace TBHVAC Controllable Pitch Exhaust Fans with Adjustable (Fixed) Pitch Fans; Revision 0
- EDP-36674; Turbine Building HVAC – Exhaust Plenum Modifications; 07/24/2012
- Evaluation 090531-01; Turbine Building Ventilation (TBHVAC), System U4100; 01/23/2011
- Evaluation 091227-01; TBHVAC, System U4100; 01/23/2011
- Evaluation 100307-01; TBHVAC, System U4100; 01/23/2011
- Evaluation 101113-01; TBHVAC, System U4100; 01/23/2011
- Evaluation 110214-01; TBHVAC, System U4100; 02/18/2011
- Evaluation 110318-01; TBHVAC System; 03/18/2011
- Evaluation 111129-01; TBHVAC System; 12/14/2011
- Evaluation 120329-01; TBHVAC System; 04/10/2012
- Evaluation 120415-01; TBHVAC System; 04/23/2012
- Evaluation 120619-01; TBHVAC System; 07/11/2012
- ODMI 10-008; Operation of the South TBHVAC Exhaust Fan; Revision D
- Root Cause Analysis Report, CARD 10-22001, Failure of Center TBHVAC Exhaust Fan; 12/15/2012
- System Health Fermi 2; TBHVAC; 1st, 2nd, 3rd, 4th Quarter, 2011
- SH-IC-331-2103-001; U41 TBHVAC System; Revision 0
- SOE No. 11-03; Obtain System Operating Data from the TBHVAC Center Exhaust Fan; Revision 0
- SOE No. 11-04; Obtain System Operating Data from the TBHVAC North and South Exhaust Fans; Revision 0
- Technical Evaluation TE-U41-12-050; TBHVAC, Two Train Operation Evaluation; Revision A

1R13 – Maintenance Risk Assessments and Emergent Work Control

- CARD 12-25709; POD Issued before Defense-in-Depth Review by SRAs; 07/02/2012
- CARD 12-27126; Further Reduction in North Reactor Feed Pump Moisture Level; 08/28/2012
- Forged Outage 12-02 Daily Report; 06/27/2012, and 07/02/2012
- Forced Outage Managers' Report FO 12-02
- Fermi Control Room Log, Unit 2; 07/05/2012; 09/04/2012
- Fermi 2 Plan of the Day; 07/02, 03, 05, 28, 30, 31, 08/01-03, 10, 13-17, 31, 09/04-07, 09/12/ and 09/14-16/2012
- Fermi 2 Forced Outage Plan
- Scheduled Risk Profile Summary; Week of 09/03/2012
- Scheduler's Evaluation for Fermi 2; 07/31/ 2012 to 8/3/2012; 09/04-14/2012
- T+1 Performance Analysis Review; 09/10-16/2012

1R15 – Operability Evaluations

- CARD 04-23820-02; General Electric to Evaluate Inspection Results from RF-10; 01/10/2006
- CARD 12-24692; North RFPT Lube Oil Has High Moisture; 05/24/2012
- CARD 12-26703; P4400F625B EECW Division 2 M/U Pump Discharge Valve Failed to Seat during Surveillance 24.208.03; 08/09/2012
- CARD 12-26834; Unable to Complete the Required Division EECW Comprehensive Pump Testing; 08/14/2012
- CARD 12-27170; North RFPT Purifier DP is Approaching Upper Limit; 08/29
- Drawing 6M721-5729-2; Emergency Equipment Cooling Water (Division II); Revision AW
- EFA-P44-12-003; Division I EECW Makeup Pump – Late Surveillance; Revisions 0, A
- NRC Letter, Elliott to Technical Specifications Task Force; 07/06/2012
- ODMI 12-005; Extended Plant Operation with Only the North RFPT; Revision 0
- Open CDM, Measurements – Sample Point, Analysis, Sample Date; 05/01/2012 – 06/21/2012
- TE-B11-12-057; Reactor Vessel Internals Assessment for Extended Plant Operation at Less Than 70 Percent Power with a Single Reactor Feed Pump in Operation; Revsion 0
- TE-B21-12-056; Evaluation of Feedwater Check Valves with Extended Plant Operation at a Reactor Power Level of Less than 70 Percent; Revision 0
- TE-J11-12-059; Long-Term Reactor Operation at 70 Percent Power with a Single Feedwater Pump; 07/13/2012

1R18 – Plant Modifications

- 10CFR50.54(q) Screen 2012-40S; 03/21/2012
- 50.59 Screen No. 12-0172; Operation at Reduced Power with the South RFP and RFPT Out of Service; Revision 0
- Drawing 6M721-5714-1; Condensate System; Revision AE
- Drawing 6M721-5715-1; Reactor Feedwater System; Revision AT
- Drawing 6M721-5717-1; Main and Reheat Steam System; Revision BN
- Drawing 6M721-5717-2; Main Turbine Extraction Steam System; Revision U
- Drawing 6M721-5717-5A; Steam Leads and Turbine Drips and Drains; Revision G
- Drawing 6M721-5717-6; Gland Steam Sealing System; Revision T
- Drawing I-2174-01; Control Center Computer Room with Raised Floor
- NRC Contact Form; C Wolfe / B Keck; Manager Projects / Manager PSE; 08/13/2012
- RERP Plan Implementing Procedure EDP-301-01; Technical Support Center; Revision 21
- EDP-35633; Index Item No. 004; Revision 0
- EDP-35633; Index Item No. B008, Document to be revised: DC5508 Volume 1; Revision 0
- EDP-35633; Index Item No. B009, Document to be revised: DC-4321 Volume 1; Revision 0
- EDP-35633; Index Item No. B
- EDP-36982; South Reactor Feed Pump/Turbine Mechanical Isolations; Revisions B and D
- EDP-36984; Abandon South Reactor Feed Pump Turbine I&C Electrical Isolations; Revisions B, D and E

1R19 – Post-Maintenance Testing

- CARD 12-25719; IPCS Core Switch Replacement; 07/02/2012
- CARD 12-25721; Unplanned Loss of IPCS Host Computers; 07/03/2012
- CARD 12-25769; Work Order 33931602 Documentation Standards Not Met; 07/05/2012
- CARD 12-25884; NQA – Evaluate Performance of Vendor Performing Radiography; 07/11/2012

- CARD 12-26184; Blown Fuses for E4150F002 While Placing HPCI in Standby IAW 23.202 and 22.000.02; 07/23/2012
- CARD 12-26967; IPCS MUX Failures; 08/16/2012
- Fermi 2 Archived Operator Log; 07/02-03/2012
- Foreign Material Control Log; Work Order 34826050, Drywell; 07/24/2012
- Foreign Material Control Log; Work Order 34828376, Drywell, 07/24/2012
- Post Maintenance Testing for WO#'s 34471516; Perform 57.000.15, Recirculating System, Performance Data Collection and Speed Limiter Setpoint Determination
- Procedure 23.425.01; Primary Containment Procedures; Revision 68
- Troubleshooting Plan for CARD 12-26968, WO 35187817; 08/23/2012
- WO 33798851; Revised PMT Step to Verify Proper EDG 14 Operation Instead of Filter Indication; 07/02/2012
- WO 33931602; EDP-35633 IPCS Core Switch Replacement in CRB; 01/25/2012
- WO 33931615; EDP-35633 IPCS Switch Post Modification Testing; 01/25/2012
- WO 34379151; EDP-35633 Electrical Power Changes in Control Room Balcony (Tie ins); 04/25/2012
- WO 34471516; Troubleshoot/Repair Reactor Recirculating B MG Set Speed Oscillations Noted by Panel Operators, Reactor pow; 05/13/2012
- WO 34718804; EDP-36982 – Mechanical Isolation on 12 IN Start Up Bypass Pipe; 07/02/2012
- WO 34718834; EDP-36982 – Mechanical Isolation on 24IN Discharge to S. Header Pipe; 07/02/2012
- WO 34826050; Replace Failed Motor in E4150F002; 07/24/2012
- WO 34910165; Install/Remove Monitor Equipment for RRMG Set A; 07/26/2012
- WO 35317095; Inspect Breaker 65G-G3 for the NORTH RRMG SET Drive Motor; 09/17/2012

1R20 - Outage Activities

- ANSI N 14.6-1978; Standard for Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials; 1978
- CARD 06-20368; Evaluate the Use of Carousel Nut Rack; 01/26/2006
- CARD 12-26252; B3105F031A Went Open While North RRMG Set Field Breaker was being Racked Out; 07/25/2012
- CARD 12-26356; IRM E Fails Downscale in Range 10; 07/28/2012
- CARD 12-26360; 4D4 'Turbine Trip Protection Fault' in alarm; 07/29/2012
- CARD 12-26362; Main Turbine Speed/Load Demand Indicator not Indicating Properly; 07/29/2012
- CARD 12-27259; NRC Identified – Revise Calculation DC-4381, Volume I; 08/31/2012
- CARD 12-27276; NRC Identified – Revise Calculation DC-3266, Volume I; 08/31/2012
- CARD 12-27498; NRC Question – RPV Head Lifting Lug Qualification; 09/11/2012
- CARD 12-27537; NRC Identified NDE Inspections; 09/12/2012
- CARD 12-27559; NRC Questions – GE's Qualification of RPV Head Strongback' 12/13/2012
- Calculation DC-3266; Design Check for Rigging Apparatus; Revision A
- Calculation DC-4381; Modified Lugs for Drywell Head to Accommodate Strongback Design; Revision 0
- Crane and Heavy Lift Inspection (OpESS FY2007-03)
- Condition Reports Initiated as a Result of NRC Inspection (OpESS FY 2007-03)
- Condition Report Reviewed during NRC Inspection (OpESS FY 2007-03)
- Design Specification 3071-375; Heavy Load Rigging Specification; Revision C
- Drawing 6C721-2201; Design and Details of Dryer/Separator Lifting Device; Revision B
- Drawing 6C721-2202; Modification to Hook Box Slings and Turnbuckles for Dryer/Separator Lifting Device; Revision A

- Drawing 6C721-2803; Reactor Building 5th Floor Heavy Load Analysis Travel Pathway; Revision A
- General Electric Calculation 0000-0051-8262-1; Reactor Pressure Vessel Strongback, 2006
- General Electric Drawing 124D1045; Dryer/Separator Slings Wet Transfer; Revision 0
- General Electric Drawing 131C7765; Outline Head Strongback; Revision 0
- General Electric Drawing 730-179; Head Strongback; Revision 5
- General Electric Drawing 796E727; Modification Head Strongback, Revisions 1 and 2
- General Electric Design Record File F12-00039; Reactor Pressure Vessel Strongback, 1984
- Letter, NRC to Detroit Edison; Control of Heavy Loads at Fermi 2 in Accordance with NUREG-0612; 11/01/1983
- Letter, NRC to Detroit Edison, Issuance of Supplement No. 5 to NUREG-0798-Fermi 2; 03/21/1985
- NUREG-0798 Supplement 5; Safety Evaluations Report Related to the Operation of Fermi 2; March 1985
- Nuclear Generation Memorandum File 0801.05; Testing of Special Lifting Devices; 12/11/1990
- Nuclear Quality Assurance Audit Report 11-0109; Quality Assurance Audit of Maintenance Programs; 08/18/2011
- Preventative Maintenance Activity F130; Perform NDE on Special Lifting Device
- Procedure 22.000.02; Plant Startup to 25 Percent Power; Revision 84
- Procedure 22.000.03; Power Operation 25 Percent to 100 Percent to 25 Percent; Revision 88
- Procedure 23.138.01; Reactor Recirculation System; Revision 106
- Procedure 32.717.01; Reactor Building Crane Operation; Revision 6
- Procedure 32.RIG.018; Guidelines and Practices for the Use of Hoisting and Rigging Equipment; Revision 8
- Procedure 35.710.025; Reactor Vessel Disassembly; Revision 17
- Procedure 35.710.026; Reactor Vessel Reassembly; Revision 14
- Procedure 35.717.003; Reactor Building Crane – Frequent and Periodic Inspection; Revision 7
- Procedure 39.NDE.002; Magnetic Particle Examination by the AC/DC Yoke Method, Enclosure D; Revision 25
- Self-Assessment of Fermi's Rigging, Lifting, and Material Handling Program; 08/31/2011
- Summary Report 0000-0051-8262-1-S; RPV Head Strongback Evaluation Summary Heavy Loads Design Adequacy Evaluation; 03/16/2006
- Technical Service Request 32431; Alternate Rigging Apparatus for the RPV Dryer and Separator; 03/28/2003
- WO 28254523; NDE Inspection of Welds and Critical Areas. Defining Critical Areas as the Outer Perimeter of Critical PM; 03/28/2009
- WO 31670200; Conduct Preparation for Refueling Inspection(s); 03/28/2012
- WO 31675298; Conduct Preparation for Refueling Inspection(s); 03/14/2012
- WO 31790854; Perform Frequent Inspection and Functionality Test Reactor Building Overhead Crane per 35.717.003; 03/18/2012

<u>1R22 – Surveillance Testing</u>

- Foreign Material Control Log; Work Order 34826050, Drywell; 07/24/2012
- Foreign Material Control Log; Work Order 34828376, Drywell, 07/24/2012
- Procedure 23.425.01; Primary Containment Procedures; Revision 68
- Procedure 24.202.08; HPCI Time Response and Pump Operability Test at 1025 PSI; Revision 5
- Risk Management Plan for the performance of 24.321.07 (72CF throwover test); 08/13/2012
- Temporary Change Notice 12311; 24.202.08, HPCI Time Response and Pump Operability Test at 1025 PSI; 08/27/2012

- WO 30757211; Perform 43.401.206 Section 6.1 and 6.2 LLRT for Airlock (X-2) (F.O.) 07/25/2012
- WO 32541096; Perform 24.202.08 Sec-5.2, HPCI Pump LSFT and Operability Test at 1025 PSIG; 08/30/2012
- WO 33270809; Perform 24.307.17, Section 5.1, EDG 14 Start and Load Test – Slow Start; 07/02/2012
  WO 33761950; Perform 24.321.07, 480V Swing Bus 72 CF Automatic Throwover Scheme Operability; 08/13/2012

<u>1EP6 - Drill Evaluation</u>

- Drill Package Scenario 51; August 30, 2012

<u>2RS1 - Radiological Hazard Assessment and Exposure Controls</u> (71124.01)

- CARD 12-22833; Two Individuals Entered A High Radiation Area In The Torus Room Without Being Briefed On Their Radiological Conditions; dated April 6, 2012
- Fermi 2 Radiation Protection Conduct Manual; MRP06; Accessing High Radiation, Locked High Radiation, and Very High Radiation Areas at Fermi 2; Revision 12
- Fermi 2 Radiation Protection Conduct Manual; MRP15; Controlling Radioactive Material Outside the Plant Radiologically Restricted Area (RRA); Revision 11
- Fermi 2 Radiation Protection Conduct Manual; MRP16; Use of Onsite Storage Facility; Revision 6
- Fermi 2 Radiation Protection Conduct Manual; MRP31; Control of Keys for High Radiation, Locked High Radiation, and Very High Radiation Areas at Fermi 2 Including Storage and Inventory of Fermi 1 Keys; Revision 1
- Plant Technical Procedure – Fermi 2; Radiation Protection Procedure; 67.000.100; Performing Surveys and Monitoring Work; Revision 39
- Plant Technical Procedure – Fermi 2; Radiation Protection Procedure; 67.000.101; Posting and Deposting of Radiological Hazards; Revision 22
- Radiological Engineering – Fermi 2; Work Instruction; National Source Tracking System; WI-RE-013; Revision 0

<u>2RS2 - Occupational ALARA Planning and Controls</u> (71124.02)

- CARD 12-24384; RF15 Radiation Exposure; dated August 28, 2012
- Chemistry Operating Standard; COS 007; Fermi 2 Strategic Water Chemistry Plan; dated August 20, 2009
- Fermi 2; Radiation Protection Conduct Manual; MRP02; Administrative Controls; Revision 16
- Fermi 2 Radiation Protection Conduct Manual; MRP05; ALARA/RWPs; Revision 8
- Fermi 2; Work Control Conduct Manual; MWC07; Online Scheduling Process; Revision 13
- Fermi 2; Work Control Conduct Manual; MWC10; Work Package Preparation; Revision 21
- Fermi 2; Work Control Conduct Manual; MWC15; Elevated Risk Management; Revision 11
- Nuclear Quality Assurance Surveillance Report 12-1002; RF 15 Outage Preparations; dates conducted January 30 to February 24, 2012
- Plant Technical Procedure – Fermi 2; 63.000.100; ALARA Procedure; Revision 38
- Plant Technical Procedure – Fermi 2; 63.000.200; ALARA Reviews; Revision 32
- Quick Hit Self Assessment: Operational ALARA Planning and Controls and Radiological Hazard Assessment and Exposure Controls; dated February 24, 2012
- Quick Hit Self Assessment: Operational ALARA Planning and Controls/Performance Indicator Verification – July 2012; dated August 13, 2012

- Radiation Protection Operation – Fermi 2; Work Instruction for Area and Installed Radiation Monitors for Pre-Job Planning; Revision 4
- RF15 – M24; Water Movement Plan; dated November 22, 2011
- RFO15 ALARA Post-Outage Review; March 26, 2012, to May 5, 2012
- RWP 1230225; Drywell Access Above 627' During Movements of Irradiated Fuel or Components; Revision 03
- RWP 125001; Perform Refuel Activities on RB-5; various revisions

4OA1 – Performance Indicator Verification

- CARD 11-25029; RCIC Pump Suction Pressure High, ARP 1D73; 05/17/1011
- Electronic Dosimetry Dose and Dose Rate Alarm Logs; various dates 2011 and 2012
- FBB-60; Fermi-2 Business Plan Performance Indicators; Revision 26
- Fermi 2 Archived Operator Log; 06/01/2010 to 07/01/2010
- MSPI Derivation Report; MSPI Emergency AC Power System; 07/23/2012
- MSPI Derivation Report; MSPI Heat Removal System; 07/23/2012
- MSPI Derivation Report; MSPI High Pressure Injection System; 07/13/2012
- MSPI Indicator Margin Remaining in Green, Fermi Unit 2, Period Ending June 2012
- Plant Technical Procedure – Fermi 2; Chemistry Operating Procedure; 73.714.01; Plant Process Sampling P33-P405A; Reactor Building Sample Panel; Revision 3
- Plant Technical Procedure – Fermi 2; Chemistry Surveillance; 74.000.19; Chemistry Routine Surveillances; Revision 24
- Plant Technical Procedure – Fermi 2; Radiochemistry Procedure; 76.000.05; Operation of Chemistry Gamma Spectroscopy Systems; Revision 15
- Plant Technical Procedure – Fermi 2; Radiochemistry Procedure; 76.000.34; Reactor Coolant Analysis; Revision 10
- Radiation Protection – Fermi 2; Work Instruction for INPO CDE Data; WI-RP-009; Revision 2

4OA2 – Identification and Resolution of Problems

- CARD 10-29509; Control Rod 10-35 Did Not Fully Insert During Scram; 10/25/2012
- CARD 11-20585; Need to Establish Supplemental Biennial Surveillance Testing; 01/19/2011
- CARD 11-30357; HCU 10-35; 11/18/2011
- CARD 12-24565; MG Set Stops Incorrectly Set During Performance of 54.000.20; 05/17/2012
- CARD 12-26429; IST Program CPT for P4400C002A/B; 07/31/2012
- CARD 12-26834; Unable to Complete the Required Division I EECW CPT; 08/14/2012
- CARD 12-26860; Determine Applicability TS 3.0.2 Restriction to Initial Performance of EECW CPT; 09/05/2012
- CARD 12-27426; NRC Concern – Significance Level Determination for CARD 11-30357; 09/07/2012
- Enforcement Guidance Memorandum 12-001; 02/24/2012
- Engineering Support Conduct Manual Chapter 23; Inservice Inspection and Testing; Revision 17
- Inservice Testing Program Third Ten-Year Interval; 10/10/2011
- Program Health Report Fermi 2; InService Testing Program; Year 2012, 1st and 2nd Quarter; Year 2011, 3rd and 4th Quarter; Year 2010, 3rd and 4th Quarter
- Root Cause Evaluation for 11-30357, "HCU 10-35"; 06/26/2012
- Quality Assurance Conduct Manual, Chapter 11; Condition Assessment Resolution Document; Revision 35

- WO 32480830; Perform 54.000.20 Reactor Recirculation system MG Set Scoop Tube Positioner Operability; 05/16/2012

4OA3 - Follow-Up of Events and Notices of Enforcement Discretion

- Estimated Critical Position Review and Approval Checklist; 09/17/2012
- Post-Scram Data and Evaluation; CARD 12-27639; 09/14/2012
- Procedure 22.000.02; Plant Startup to 25% Power; Revision 84

4OA5 - Other Activities

- CARD 12-27314; Blockage experienced in floor drains; 09/04/2012

## LIST OF ACRONYMS USED

| | |
|---|---|
| AC | Alternating Current |
| ADAMS | Agencywide Document Access Management System |
| ALARA | As-Low-As-Is-Reasonably-Achievable |
| ANSI | American National Standards Institute |
| ASME | American Society of Mechanical Engineers |
| BWR | Boiling Water Reactor |
| CARD | Corrective Action and Resolution Document |
| CFR | Code of Federal Regulations |
| CPT | Comprehensive Pump Test |
| DCS | Distributed Control System |
| DRP | Division of Reactor Projects |
| ECCS | Emergency Core Cooling System |
| EDG | Emergency Diesel Generator |
| EDP | Engineering Design Package |
| EECW | Emergency Equipment Cooling Water |
| EGM | Enforcement Guidance Memorandum |
| HPCI | High Pressure Coolant Injection |
| HRA | High Radiation Area |
| IEL | Initiating Event Likelihood |
| IMC | Inspection Manual Chapter |
| IP | Inspection Procedure |
| IST | Inservice Testing |
| kV | Kilovolt |
| LER | Licensee Event Report |
| LORHR | Loss of Residual Heat Removal |
| MG | Motor Generator |
| MSPI | Mitigating Systems Performance Index |
| NCV | Non-Cited Violation |
| NEI | Nuclear Energy Institute |
| NRC | U.S. Nuclear Regulatory Commission |
| ODCM | Offsite Dose Calculation Manual |
| ODMI | Operational Decision Making Issue |
| OM | Operations and Maintenance |
| OpESS | Operating Experience Smart Sample |
| PARS | Publicly Available Records System |
| PI | Performance Indicator |
| PMT | Post-Maintenance Testing |
| POS | Plant Operating State |
| PWR | Pressurized Water Reactor |
| RCIC | Reactor Core Isolation Cooling |
| RCS | Reactor Coolant System |
| RFO | Refueling Outage |
| RFP | Reactor Feed Pump |
| RFPT | Reactor Feed Pump/Turbine |
| RHR | Residual Heat Removal |
| RPV | Reactor Pressure Vessel |
| RRMG | Reactor Recirculation Motor Generator |
| SDP | Significance Determination Process |

| | |
|---|---|
| SOE | Significant Operation or Evolution |
| SRA | Senior Reactor Analyst |
| TS | Technical Specification |
| UFSAR | Updated Final Safety Analysis Report |
| URI | Unresolved Item |
| VHRA | Very High Radiation Area |
| WO | Work Order |

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 Agencywide Document Access and Management 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/*

Jamnes L. Cameron, Chief
Branch 6
Division of Reactor Projects

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

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w/Attachment: Supplemental Information

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|---|---|---|---|---|---|---|---|---|
| NAME | JRutkowski:dtp<br>*TCB for | | JCameron | | | | | |
| DATE | 10/31/12 | | 10/31/12 | | | | | |

***OFFICIAL RECORD COPY***

Letter to J. Plona from J. Cameron dated October 31, 2012.

SUBJECT: FERMI POWER PLANT, UNIT 2, INTEGRATED INSPECTION REPORT 05000341/2012004

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