UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE RD. SUITE 210
LISLE, ILLINOIS 60532-4352

February 13, 2019

Keith Polson Senior VP
and Chief Nuclear Officer
DTE Energy Company
Fermi 2 – 260 TAC
6400 North Dixie Highway
Newport, MI 48166

SUBJECT: FERMI POWER PLANT, UNIT 2—NRC INTEGRATED INSPECTION REPORT 05000341/2018004

Dear Mr. Polson:

On December 31, 2018, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Fermi Power Plant, Unit 2. On January 14, 2018, the NRC inspectors discussed the results of this inspection with you and other members of your staff. The results of this inspection are documented in the enclosed report.

Based on the results of this inspection, the NRC has identified five issues that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has also determined that three violations are associated with these issues. Because condition reports were generated to address these issues, these violations are being treated as Non-Cited Violations (NCVs), consistent with Section 2.3.2 of the Enforcement Policy. These NCVs are described in the subject inspection report.

If you contest the violations or significance of these NCVs, 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 copies to the Regional Administrator, Region III; the Director, Office of Enforcement; and the NRC Resident Inspector at the Fermi Power Plant.

If you disagree with a cross-cutting aspect assignment or a finding not associated with a regulatory requirement 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 U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555–0001; with copies to the Regional Administrator, Region III; and the NRC resident inspector at the Fermi Power Plant.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

/RA/

Eric Duncan, Chief
Branch 4
Division of Reactor Projects

Docket No. 50–341
License No. NPF–43

Enclosure:
Inspection Report 05000341/2018004

cc: Distribution via LISTSERV®

Letter to Keith Polson from Eric Duncan dated February 13, 2019

SUBJECT: FERMI POWER PLANT, UNIT 2—NRC INTEGRATED INSPECTION REPORT 05000341/2018004

DISTRIBUTION:
Michael McCoppin
RidsNrrPMFermi2 Resource
RidsNrrDirsIrib Resource
Darrell Roberts
John Giessner
Jamnes Cameron
Allan Barker
DRPIII
DRSIII
ROPreports.Resource@nrc.gov

ADAMS Accession Number: ML19044A632

OFFICE	RIII							
NAME	EDuncan:bw							
DATE	2/13/2019							

OFFICIAL RECORD COPY

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Number:	50–341
License Number:	NPF–43
Report Number:	05000341/2018004
Enterprise Identifier:	I–2018–004–018
Licensee:	DTE Energy Company
Facility:	Fermi Power Plant, Unit 2
Location:	Newport, MI
Dates:	October 1, 2018 through December 31, 2018
Inspectors:	T. Briley, Senior Resident Inspector P. Smagacz, Resident Inspector T. Taylor, Resident Inspector R. Baker, Senior Operations Engineer J. Bozga, Senior Reactor Inspector A. Dahbur, Senior Reactor Inspector V. Myers, Senior Health Physicist G. Hansen, Senior Emergency Preparedness Inspector T. Ospino, Reactor Engineer
Approved by:	E. Duncan, Chief Branch 4 Division of Reactor Projects

Enclosure

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring licensee performance by conducting an integrated quarterly inspection at the Fermi Power Plant, Unit 2 in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information. Findings and violations being considered in the NRC's assessment are summarized in the table below.

List of Findings and Violations

<table>
<tr><td colspan="4">Inadequate Rigging and Lifting Practices Result in Damage to Division 2 Residual Heat Removal Structures and Components</td></tr>
<tr><td>Cornerstone</td><td>Significance</td><td>Cross-Cutting Aspect</td><td>Report Section</td></tr>
<tr><td>Mitigating Systems</td><td>Green
FIN 05000341/2018004–01
Open and Closed</td><td>[H.12] – Human Performance – Avoid Complacency</td><td>71111.13—Maintenance Risk Assessments and Emergent Work Control</td></tr>
<tr><td colspan="4">A self-revealed finding of very low safety significance was identified when licensee personnel failed to follow site standards and procedures for safe rigging and handling practices while lifting the Division 2 residual heat removal service water outlet flow control valve stem and attached disc. This resulted in a dropped load and damage to safety-related equipment. Specifically, a detailed pre-job brief was not performed and an infield decision was made to use an inappropriate hook-to-hook rigging configuration that resulted in a dropped load and subsequent damage to Division 2 residual heat removal system structures and components.</td></tr>
</table>

<table>
<tr><td colspan="4">Failure to Follow the Radiation Work Permit For Entry Into a High Radiation Area</td></tr>
<tr><td>Cornerstone</td><td>Significance</td><td>Cross-Cutting Aspect</td><td>Report Section</td></tr>
<tr><td>Occupational Radiation Safety</td><td>Green
NCV 05000341/2018004–04
Open and Closed</td><td>[H.4] – Human Performance – Teamwork</td><td>71124.01—Radiological Hazard Assessment and Exposure Controls</td></tr>
<tr><td colspan="4">A self-revealed finding of very low safety significance and an associated Non-Cited Violation (NCV) of Technical Specification (TS) 5.4.1, "Procedures," and TS 5.7.1, "High Radiation Area [HRA]," was identified when radiation workers violated Radiation Work Permit (RWP) requirements by entering a HRA without notifying Radiation Protection (RP) staff prior to entry and without being briefed for the existing radiological conditions.</td></tr>
</table>

Failure to Establish Radiological Conditions for a Locked High Radiation Area Entry			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Occupational Radiation Safety	Green NCV 05000341/2018004–05 Open and Closed	[H.10] – Human Performance – Bases For Decisions	71124.01—Radiological Hazard Assessment and Exposure Controls
A self-revealed finding of very low safety significance and an associated NCV of TS 5.7.2, “High Radiation Area,” was identified when RP staff allowed workers to enter a Locked High Radiation Area where radiological conditions had not been established.			

Failure to Verify the Adequacy of the Design of the 4160 Volt Safety Bus 64B, 64C, 65E, and 65F Degraded Voltage Protection Schemes			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000341/2018004–06 Open and Closed	None	71152—Problem Identification and Resolution
A self-revealed finding of very low safety significance and associated NCVs of 10 CFR 50, Appendix B, Criterion III, “Design Control”; TS 3.3.8.1, “Loss of Power (LOP) Instrumentation”; TS 3.5.1, “ECCS [Emergency Core Cooling System] – Operating”; and TS 3.8.1, “AC Sources – Operating,” were identified when licensee personnel failed to verify the adequacy of the design of the 4160 volt safety-related bus 64B, 64C, 65E, and 65F under voltage (degraded voltage) protection relaying circuitry to ensure the associated relaying did not operate during emergency diesel generator load sequencer operation. Specifically, during various loss of offsite power and loss of coolant accident scenarios, a load shed signal from the degraded voltage protection relaying would prevent the associated residual heat removal pump from automatically starting while the load sequencer was operating.			

Rainwater Intrusion into Degraded Metal-Clad Enclosure for Bus 1–2B Results in Partial Loss of Offsite Power, Reactor Scram, and Emergency Core Cooling System Initiation			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Green FIN 05000341/2018004–07 Open and Closed	[H.12] – Human Performance – Avoid Complacency	71153—Follow-Up of Events and Notices of Enforcement Discretion
A self-revealed finding of very low safety significance was identified when on April 14, 2018, rainwater intrusion into a degraded, metal-clad enclosure for Bus 1–2B switchgear in the Division 1 120 kilovolt switchyard caused a partial loss of offsite power, reactor scram, and ECCS initiation. Specifically, licensee personnel failed to adequately maintain the metal-clad enclosure for Bus 1–2B and did not recognize the potential vulnerability to plant operation.			

Additional Tracking Items

Type	Issue Number	Title	Report Section	Status
Unresolved Item	05000341/2018004–02	Division 2 Residual Heat Removal Service Water Outlet Flow Control Valve Stem Failures	71111.15	Open
Unresolved Item	05000341/2018004–03	Torus Coating Work Not Covered Under Work Hour Rules	71111.20	Open
Licensee Event Report	05000341/2018–002–00	Loss of Division 1 Offsite Power Causes Partial Loss of Feedwater Leading to ECCS Injection and Reactor Scram	71153	Closed

PLANT STATUS

Unit 2 began the inspection period shut down for a planned refueling outage. The reactor was restarted on October 25, 2018. The unit returned to full power on October 29, 2018. On October 30, 2018, the unit was reduced to 80 percent power for a planned rod pattern adjustment. The unit returned to full power the same day. On November 2, 2018, the unit was reduced to 60 percent power for a planned rod pattern adjustment. The unit returned to full power on November 3, 2018. On November 6, 2018, the unit was reduced to 66 percent power for a planned rod pattern adjustment. The unit returned to full power the same day. On December 3, 2018, the unit was reduced to 83 percent power to troubleshoot an electrical issue with the main generator. The unit was shut down on December 6, 2018, to further investigate and repair the electrical issue with the main generator. The unit restarted on December 31, 2018.

INSPECTION SCOPES

Inspections were conducted using the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection unless otherwise noted. Currently approved IPs with their attached revision histories are located on the public website at http://www.nrc.gov/reading-rm/doc-collections/insp-manual/inspection-procedure/index.html. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter (IMC) 2515, "Light-Water Reactor Inspection Program - Operations Phase." The inspectors performed plant status activities described in IMC 2515 Appendix D, "Plant Status" and conducted routine reviews using IP 71152, "Problem Identification and Resolution." The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel to assess licensee performance and compliance with Commission rules and regulations, license conditions, site procedures, and standards.

REACTOR SAFETY

71111.01—Adverse Weather Protection

Seasonal Extreme Weather (1 Sample)

The inspectors evaluated the licensee's readiness for seasonal extreme weather conditions prior to the onset of winter weather during the weeks ending November 17, 2018 and November 24, 2018.

External Flooding (1 Sample)

The inspectors evaluated the licensee's readiness to cope with external flooding during the weeks ending November 17, 2018 and November 24, 2018.

71111.04—Equipment Alignment

Partial Walkdown (3 Samples)

The inspectors evaluated system configurations during partial walkdowns of the following systems/trains:

(1) Division 2 core spray with shutdown cooling out of service during the week ending October 13, 2018;
(2) Division 1 nuclear instrument air system (NIAS) with Division 2 NIAS out of service for planned maintenance during the week ending November 24, 2018; and
(3) Division 1 emergency equipment cooling water (EECW) while Division 2 EECW was out of service for planned maintenance during the week ending December 1, 2018.

71111.05AQ—Fire Protection Annual/Quarterly

Quarterly Inspection (4 Samples)

The inspectors evaluated fire protection program implementation in the following selected areas:

(1) Residual heat removal (RHR) complex first floor — Division 1 pump room and emergency diesel generator 12 room during the week ending October 6, 2018;
(2) Drywell — general areas during the weeks ending October 13, 2018 and October 20, 2018;
(3) Reactor building second floor — steam tunnel area during the week ending October 20, 2018;
(4) Turbine building second floor — steam tunnel area during the week ending October 20, 2018; and
(5) Turbine building third floor — main generator area during the week ending December 22, 2018.

71111.06—Flood Protection Measures

Internal Flooding (1 Sample)

The inspectors evaluated internal flooding mitigation protection in the auxiliary building first floor, basement, and sub-basement during the week ending December 22, 2018.

Cables (1 Sample)

The inspectors evaluated cable submergence protection in Manhole 16946C during the weeks ending July 21, 2018 through October 6, 2018.

71111.07—Heat Sink Performance

Heat Sink (1 Sample)

The inspectors evaluated Division 1 RHR heat exchanger cleaning/eddy current performance during the week ending October 6, 2018.

71111.08—In-Service Inspection Activities (1 Sample)

The inspectors assessed the effectiveness of the licensee’s programs for monitoring degradation of the reactor coolant system boundary, risk-significant piping system boundaries, and the containment boundary by reviewing the following activities from September 24, 2018 to October 11, 2018:

(1) Liquid penetrant examination (PT) of 2" socket weld for standby liquid control system component identification (ID) FW C41–2979–P;
(2) Magnetic particle examination (MT) of top head lifting lug attachment welds component ID 8–319C and 8–319D;
(3) Ultrasonic examination (UT) record of core DP nozzle-to-safe end butt welds for the standby liquid control system component ID 5–315;
(4) Examination records with relevant indications accepted for continued service – reportable magnetic particle indication found on support lug SW–PS–2–A2–AA4; and
(5) Welding Work Order 34306179 replacement of emergency diesel generator 14 service water outlet valve component ID R3000F139D and Work Order 44879308 pipe replacement of Division 2 emergency equipment cooling water heat exchanger service water return piping component ID MK–P45–3353–10.

<u>71111.11—Licensed Operator Requalification Program and Licensed Operator Performance</u>

<u>Operator Requalification</u> (1 Sample)

The inspectors observed and evaluated an operations crew evaluated scenario in the plant training simulator on November 13, 2018.

<u>Operator Performance</u> (1 Sample)

The inspectors observed and evaluated reactor startup activities during the week ending October 27, 2018.

<u>Operator Exams</u> (1 Sample)

The inspectors reviewed and evaluated requalification examination results on December 18, 2018.

<u>71111.12—Maintenance Effectiveness</u>

<u>Routine Maintenance Effectiveness</u> (1 Sample)

The inspectors evaluated the effectiveness of routine maintenance activities associated with the following equipment and/or safety significant functions:

(1) High pressure coolant injection (HPCI) during the week ending December 22, 2018.

<u>71111.13—Maintenance Risk Assessments and Emergent Work Control</u> (4 Samples)

The inspectors evaluated the risk assessments for the following planned and emergent work activities:

(1) Planned torus desludging and coating repairs during the week ending October 12, 2018;
(2) Planned reactor head lift and installation on the reactor vessel during the week ending October 20, 2018;
(3) Planned Division 2 residual heat removal service water (RHRSW) outlet flow control valve E1150–F068B stem and disc installation during the weeks ending December 1, 2018 and December 8, 2018; and

(5) Planned heavy lift of the main generator rotor during the weeks ending December 15, 2018 and December 22, 2018.

71111.15—Operability Determinations and Functionality Assessments (8 Samples)

The inspectors evaluated the following operability determinations and functionality assessments:

(1) Operability and functionality of control room recirculation loop jet pump flow indication following intermittent indications, as documented in corrective action resolution document (CARD) 18–25563;
(2) Operability and functionality of Division 2 core spray following identification of packing leakage on Division 2 core spray inboard isolation valve E2150–F005B, as documented in CARD 18–28405;
(3) Operability and functionality of Division 1 ultimate heat sink following identification of a through-wall pipe leak, as documented in CARD 18–28624;
(4) Operability and functionality of HPCI following identification of packing leakage on outboard steam admission isolation valve E4150–F003, as documented in CARD 18–28795;
(5) Operability and functionality of Division 2 RHR heat exchanger following damage to structural support beams as documented in CARDs 18–29568 and 18–29583;
(6) Operability and functionality of the seismic monitoring system following a failed surveillance test, as documented in CARD 18–26407;
(7) Operability and functionality of the main steam isolation valves following various local leak rate testing failures, as documented in CARDs 18–27189, 18–27188, and 18–27165; and
(8) Operability and functionality of Division 2 RHRSW following the stem failure of outlet flow control valve E1150–F068B, as documented in CARD 18–29027.

71111.18—Plant Modifications (2 Samples)

The inspectors evaluated the following temporary or permanent plant modifications:

(1) Division 2 RHRSW outlet flow control valve E1150–F068B valve internals removal, as documented in temporary modification package 18–0030; and
(2) 4160 volt safety bus 64B, 64C, 65E, and 65F degraded voltage protection scheme changes, as documented in permanent modification package 80065.

71111.19—Post Maintenance Testing (9 Samples)

The inspectors evaluated the following post maintenance tests:

(1) Division 2 RHRSW outlet flow control valve E1150–F068B bonnet replacement during the weeks ending September 29, 2018 and October 6, 2018;
(2) Emergency diesel generator 11 and 12 service water operability tests following emergency diesel generator 11 and 12 service water piping replacement during the weeks ending October 13, 2018 and October 20, 2018;
(3) Control rod friction testing following replacement of multiple control rods during the week ending October 20, 2018;
(4) Reactor vessel hydrostatic testing following reactor vessel reassembly during the week ending October 20, 2018;

(5) Fuel bundle core location verification following multiple fuel bundle replacements and core shuffle during the week ending October 20, 2018;
(6) Feedwater check valve B2100–F076A and B2100–F076B local leak rate tests following check valve seat replacements during the week ending October 20, 2018;
(7) Division 2 core spray inboard isolation valve E2150–F005B local leak rate test following stem replacement and packing leakage during the week ending October 20, 2018;
(8) Main steam isolation valve local leak rate testing following repairs to B outboard (B2103–F028B) and D inboard (B2103–F022D) main steam isolation valves during the weeks ending October 20, 2018; and
(9) Division 2 RHRSW test following outlet flow control valve E1150–F068B stem failure and temporary modification 18–0030 to remove valve internals during the weeks ending November 10, 2018 and November 17, 2018.

<u>71111.20—Refueling and Other Outage Activities</u> (1 Complete Sample and 1 Partial Sample)

The inspectors evaluated refueling outage 19 activities from September 22, 2018 through October 26, 2018.

The inspectors evaluated forced outage activities due to an electrical issue with the main generator from December 6, 2018 to December 31, 2018. The inspectors completed all applicable inspection procedure sections with the exception of 03.01.b and 03.01.e. This constituted a partial sample.

<u>71111.22—Surveillance Testing</u>

The inspectors evaluated the following surveillance tests:

<u>Routine</u> (4 Samples)

(1) Division 2 ECCS core spray logic functional testing during the week ending October 6, 2018;
(2) Division 2 18–month 130/260 volt direct current battery testing during the week ending October 6, 2018;
(3) Simultaneous ECCS start of all four emergency diesel generators during the week ending October 20, 2018; and
(4) Emergency diesel generator 11 loss of offsite power (LOP) and ECCS start with LOP testing during the week ending October 20, 2018.

<u>In-Service</u> (1 Sample)

(1) Division 2 core spray check valve E2150–F006B testing during the week ending October 6, 2018.

<u>Containment Isolation Valve</u> (1 Sample)

(1) Division 2 core spray inboard isolation valve E2150–F005B local leak rate test during the week ending October 6, 2018.

71114.04—Emergency Action Level and Emergency Plan Changes (1 Sample)

The inspectors completed an evaluation of submitted Emergency Action Level and Emergency Plan changes on November 1, 2018. This evaluation does not constitute NRC approval of the change.

RADIATION SAFETY

71124.01—Radiological Hazard Assessment and Exposure Controls

Radiological Hazard Assessment (1 Sample)

The inspectors evaluated radiological hazards assessments and controls.

Instructions to Workers (1 Sample)

The inspectors evaluated worker instructions.

Contamination and Radioactive Material Control (1 Sample)

The inspectors evaluated contamination and radioactive material controls.

Radiological Hazards Control and Work Coverage (1 Sample)

The inspectors evaluated radiological hazards control and work coverage.

High Radiation Area and Very High Radiation Area Controls (1 Sample)

The inspectors evaluated risk-significant high radiation area and very high radiation area controls.

Radiation Worker Performance and Radiation Protection Technician Proficiency (1 Sample)

The inspectors evaluated radiation worker performance and radiation protection technician proficiency.

71124.02—Occupational As Low As Reasonably Achievable Planning and Controls

Implementation of As Low As Reasonably Achievable and Radiological Work Controls (1 Sample)

The inspectors reviewed as low as reasonably achievable (ALARA) practices and radiological work controls by reviewing the following activities:

(1) Radiation Work Permit (RWP) 184001; Reactor Building Fifth Floor — reactor disassembly and support activities;
(2) RWP 183016; Main Steam Isolation Valves — drywell & reactor building steam tunnel — maintenance and inspection; and
(3) RWP 182025; G33 (Reactor Water Cleanup), P73 (Hydrogen Water Chemistry (HWC)) system maintenance and inspection.

Radiation Worker Performance (1 Sample)

The inspectors evaluated radiation worker and radiation protection technician performance.

OTHER ACTIVITIES – BASELINE

71151—Performance Indicator Verification (3 Samples)

The inspectors verified the licensee performance indicators submittals listed below:

(1) BI01: Reactor Coolant System (RCS) Specific Activity Sample —1 Sample (October 1, 2017 – June 30, 2018);
(2) OR01: Occupational Exposure Control Effectiveness —1 Sample (October 1, 2017 – June 30, 2018); and
(3) PR01: Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual (RETS/ODCM) Radiological Effluent Occurrences — 1 Sample (October 1, 2017 – June 30, 2018).

71152—Problem Identification and Resolution

Semiannual Trend Review (1 Sample)

The inspectors reviewed the licensee's corrective action program for trends in the following area that might be indicative of a more significant safety issue.

(1) Operability determination program implementation.

Annual Follow-Up of Selected Issues (1 Sample)

The inspectors reviewed the licensee's implementation of its corrective action program related to the following issue:

(1) 4160 volt safety bus 64B, 64C, 65E, and 65F degraded voltage protection scheme operates during emergency diesel generator load sequencer operation, as documented in CARD 18–28990.

71153—Follow-Up of Events and Notices of Enforcement Discretion

Licensee Event Reports (1 Sample)

The inspectors evaluated the following licensee event reports (LERs) which can be accessed at https://lersearch.inl.gov/LERSearchCriteria.aspx:

(1) LER 05000341/2018–002–00 — Loss of Division 1 Offsite Power Causes Partial Loss of Feedwater Leading to ECCS Injection and Reactor Scram. A finding is documented in Inspection Results Section 71153 of this report. This LER is closed.

INSPECTION RESULTS

71111.13—Maintenance Risk Assessments and Emergent Work Control

Inadequate Rigging and Lifting Practices Result in Damage to Division 2 Residual Heat Removal Structures and Components			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green FIN 05000341/2018004–01 Open and Closed	[H.12] – Human Performance — Avoid Complacency	71111.13 — Maintenance Risk Assessments and Emergent Work Control
A self-revealed finding of very low safety significance was identified when licensee personnel failed to follow site standards and procedures for safe rigging and handling practices while lifting the Division 2 residual heat removal service water (RHRSW) outlet flow control valve stem and attached disc. This resulted in a dropped load and damage to plant safety-related equipment. Specifically, a detailed pre-job brief was not performed and an infield decision was made to use an inappropriate hook-to-hook rigging configuration that resulted in a dropped load and subsequent damage to Division 2 residual heat removal (RHR) system structures and components.			

Description:

On November 29, 2018, mechanical maintenance personnel were in the process of rigging and lifting a replacement valve stem and disc assembly for Division 2 RHRSW outlet flow control valve E1150–F068B when the valve stem and disc assembly unexpectedly came unhooked. The stem and disc assembly dropped about 15 feet to the grating in the Division 2 RHR heat exchanger room in the reactor building.

The 600 pound valve stem and disc assembly struck and damaged the following structures and components:

- Division 2 RHR pipe support snubber E11–3151–G17;
- Multiple Division 2 RHR heat exchanger structural support beams; and
- Division 2 RHR heat exchanger motor–operated bypass valve E1150–F048B limit switch cover.

The Division 2 RHRSW system was out of service for corrective maintenance when the event occurred. The Division 2 RHR pipe support snubber was declared inoperable in accordance with Technical Specification (TS) 3.0.8, "Snubbers." The Division 2 RHR system was declared inoperable in accordance with TS 3.5.1, "Emergency Core Cooling System – Operating," as a result of damage to the heat exchanger structural supports and motor–operated bypass valve.

The licensee performed an evaluation of the damaged components. A prompt operability assessment determined that although the structural integrity of the Division 2 RHR heat exchanger remained within allowable American Institute of Steel Construction (AISC) structural design specification limits, Fermi structural calculation limits were exceeded and, as a result, the Division 2 RHR system was declared operable, but non-conforming. The licensee also determined the damaged snubber required replacement along with the motor-operated bypass valve limit switch cover. Following repairs, all of the associated TS limiting conditions for operation (LCOs) were exited before their completion times were exceeded.

The licensee performed an investigation and determined the mechanical maintenance personnel involved in the rigging and lifting evolution did not perform a detailed pre-job brief required by procedure MGA 24, "Human Performance Program and Field Worker Tools;" procedure MDI–055, "Fundamental of Maintenance"; and site expectations to discuss roles and responsibilities, rigging configuration and travel path, and stop work criteria.

When mechanical maintenance personnel were challenged during transition of the load over an RHR piping interference utilizing a single chain fall hook, they made an inappropriate infield decision to connect two chain fall hooks "hook-to-hook" to pull the load over the piping interference. They did not connect each hook independently to the valve stem eyebolt as required by site lifting and rigging standard 32.RIG.18, "Guidelines and Practices for the Use of Hoisting and Rigging Equipment," since the valve stem eyebolt was too small to accommodate both chain fall hooks simultaneously. As the valve stem and disc were moved at an angle over the RHR piping, the 600 pound load shifted to the primary hook, which was the only hook attached directly to the eyebolt, and the safety link failed due to the weight of the load that it was never designed to carry. The load dropped because the other chain fall hook was not directly connected to the valve stem eyebolt.

Corrective Actions: A maintenance department stand down was held to discuss the event and associated errors. A stem and disc recovery plan was also developed utilizing additional controls intended for heavy lifts in accordance with NUREG–0612, "Control of Heavy Loads at Nuclear Power Plants," to retrieve the dropped stem and disc assembly. The licensee also planned to utilize a formal lift plan for any future lifts in the RHR heat exchanger rooms. The Division 2 RHR pipe support snubber and motor-operated bypass valve limit switch cover were replaced, and the Division 2 RHR heat exchanger structural supports were replaced or modified to restore analytical design margin. The licensee also completed a human performance and organizational effectiveness cause evaluation.

Corrective Action References: CARDs 18–29568, 18–29569, 18–29570, and 18–29583

Performance Assessment:

Performance Deficiency: The failure to follow site standards and procedures for safe rigging and handling practices while lifting the Division 2 RHRSW outlet flow control valve stem and attached disc was a performance deficiency. This resulted in a dropped load and damage to safety-related equipment.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Human Performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the failure to follow site standards and procedures for safe rigging and handling practices while lifting a valve stem and attached disc resulted in damage to safety-related Division 2 RHR structures and components and impacted their reliability and capability to perform their intended safety functions.

Significance: The inspectors assessed the significance of the finding using Inspection Manual Chapter (IMC) 0609, Appendix A, "The Significance Determination Process (SDP) For Findings At–Power," and answered "No" to all the Section A screening questions in Exhibit 2, "Mitigating Systems Screening Questions." Therefore, the inspectors determined that the finding was of very low safety significance (i.e., Green). Cross-Cutting Aspect: The finding had a cross-cutting aspect in the Avoid Complacency component of the Human Performance cross-cutting area, which states that the licensee will recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes and that individuals will implement appropriate error reduction tools. Specifically, maintenance personnel did not recognize and plan for mistakes and the inherent risk of damage to plant equipment as they were overconfident in their ability to perform the work. [H.12]
Enforcement: The inspectors did not identify a violation of regulatory requirements associated with this finding.

71111.15—Operability Determinations and Functionality Assessments

Unresolved Item (Open)	Division 2 Residual Heat Removal Service Water Outlet Flow Control Valve Stem Failures URI 05000341/2018004–02	71111.15—Operability Determinations and Functionality Assessments
Description: During refueling outage 19, which ended on October 27, 2018, new valve bonnets and valve stems were installed for Division 1 and Division 2 RHRSW outlet flow control valves E1150–F068A and E1150–F068B, respectively. On November 2, 2018, during routine biocide treatment of the RHR reservoir, the new Division 2 RHRSW outlet flow control valve E1150–F068B stem failed (broke in two) while attempting to shut down the RHRSW system. The licensee declared Division 2 RHRSW inoperable and entered a 7 day TS LCO as required by TS 3.7.1, "Residual Heat Removal Service Water (RHRSW) System." The licensee replaced the broken valve stem with a previously used stem from Division 1 RHRSW outlet flow control valve E1150–F068A. During post-maintenance testing on November 6, 2018, the replacement valve stem also failed. Since the failure and causal analysis had not been completed, the licensee elected to implement a temporary modification to completely remove the internals (stem and attached disc) from the valve and welded a seal plate to the valve bonnet. The system was subsequently returned to service on November 8, 2018. The licensee's root cause evaluation to determine why the valve stems failed had not been completed at the end of the inspection period.		
Planned Closure Action: The inspectors plan to review the licensee's root cause evaluation to determine if a performance deficiency exists.		

Licensee Action: A temporary modification was developed to remove the Division 2 RHRSW outlet flow control valve internals until the root cause was determined and appropriate corrective actions could be implemented. Corrective Action Reference: CARD 18–29027

71111.20—Refueling and Other Outage Activities

Unresolved Item (Open)	Torus Coating Work Not Covered Under Work Hour Rules URI 05000341/2018004–03	71111.20—Refueling and Other Outage Activities
Description: During refueling outage 19, which was conducted from September 22, 2018 to October 27, 2018, coating inspections and repairs were performed inside the safety-related torus, which is a component of primary containment. One of the functions of the torus is to provide a large volume of water to supply the emergency core cooling system (ECCS). As revealed through industry operating experience, if torus coatings are improperly applied and maintained, coating delamination and plugging of ECCS suction strainers could occur. The inspectors identified that the licensee classified coating work inside the torus as uncovered work and therefore the work hours of those individuals were not tracked to ensure that 10 CFR 26, “Fitness For Duty Programs,” work hour rules were followed in accordance with procedure MGA17, “Working Hour Limits.” Planned Closure Action: The inspectors plan to review the actual work hours of torus coating workers and the associated licensee evaluation to determine if the licensee’s failure to monitor work hours in accordance with 10 CFR 26 and licensee procedures was of more than minor significance. Licensee Action: A licensee cause evaluation was planned to review the actual work hours of torus coating workers and develop corrective actions to address the misclassification of torus coating work. Corrective Action Reference: CARD 19–20217		

71124.01—Radiological Hazard Assessment and Exposure Controls

Failure To Follow The Radiation Work Permit For Entry Into a High Radiation Area			
Cornerstone	Significance	Cross–Cutting Aspect	Report Section
Occupational Radiation Safety	Green NCV 05000341/2018004–04 Closed	[H.4] — Human Performance — Teamwork	71124.01—Radiological Hazard Assessment and Exposure Controls
A self-revealed finding of very low safety significance and an associated Non-Cited Violation (NCV) of TS 5.4.1, “Procedures,” and TS 5.7.1, “High Radiation Area (HRA),” was identified when radiation workers violated Radiation Work Permit (RWP) requirements and entered a			

HRA without notifying Radiation Protection (RP) personnel prior to entry and without being briefed on the radiological conditions in the area.
Description: On September 28, 2018, two individuals received a HRA briefing for entry into the drywell using RWP 183019, "B21/E21 System Maintenance & Inspection — Drywell & RB [Reactor Building] Steam Tunnel." The RWP and the licensee's TSs required a briefing of the radiological conditions prior to entry into a HRA. After exiting the drywell, the individuals entered the steam tunnel, which was a separate HRA (having radiation levels exceeding 100 millirem (mrem)/hour) that was not included in the previous briefing. The entry into the steam tunnel was identified when one of the individuals received an electronic dosimeter high dose rate alarm. Upon receipt of the high dose rate alarm, both individuals exited the HRA and reported the receipt of the high dose rate alarm to RP staff. The licensee determined that one individual had received a HRA brief for a steam tunnel entry made several days before and believed that this was sufficient for re-entry. The other individual had not received a HRA briefing for entry into the steam tunnel. The inspectors determined that this issue was self-revealed because the lack of briefings of the radiological conditions in the steam tunnel work area was identified in response to the high dose rate alarm received by one of the workers. Corrective Actions: Corrective actions included temporarily restricting access for the individuals while the licensee investigated the event, and distributing a Human Performance Reset Briefing Sheet to site personnel that discussed the issue and reiterated to personnel the need to obtain briefings from RP staff prior to entering any HRA. Corrective Action Reference: CARD 18–27507
Performance Assessment: Performance Deficiency: The failure of workers to follow RWP requirements and the licensee's TSs by entering a HRA without notifying RP staff prior to entry and without being briefed to the radiological conditions in the area was a performance deficiency. Screening: The inspectors determined that the performance deficiency was more than minor because it was associated with the Human Performance attribute of the Occupational Radiation Safety cornerstone and adversely affected the cornerstone objective of ensuring the adequate protection of worker health and safety from exposure to radiation. Specifically, worker entry into HRAs without knowledge of the radiological conditions could lead to unintended dose. Significance: The inspectors assessed the significance of the finding using IMC 0609 Appendix C, "Occupational Radiation Safety Cornerstone." The inspectors determined that the finding was of very low safety significance (i.e., Green) because: (1) it did not involve as-low-as-reasonably-achievable planning or work controls; (2) there was no overexposure; (3) there was no substantial potential for an overexposure; and (4) the ability to assess dose was not compromised. Cross-Cutting Aspect: The finding had a cross-cutting aspect in the Teamwork component of the Human Performance cross-cutting area, which states that individuals and work groups

communicate and coordinate their activities within and across organizational boundaries to ensure nuclear safety is maintained. Specifically, the two individuals did not communicate or coordinate their steam tunnel entry with RP staff. [H.4]

Enforcement:

Violation: Technical Specification 5.4.1, "Procedures," requires, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide (RG) 1.33, Revision 2, Appendix A.

Section 7(1) of RG 1.33 addresses access controls to radiation areas, including an RWP system.

MRP04, "Accessing and Working in the Radiologically Controlled Area", Section 4.3.3(4)(a), requires that all personnel do not deviate from an RWP unless specific authorization is obtained from Radiation Protection.

RWP 183019 required all personnel to notify RP staff prior to initial entry into HRAs and that all subsequent entries or actions require an RP brief of updated radiological conditions and specific RWP actions.

Technical Specification 5.7.1, "High Radiation Area," requires, in part, that individuals entering HRAs > [greater than] 100 mrem/hr [hour] but < [less than] 1000 mrem/hr be provided with or accompanied by either:

a) A radiation monitoring device that continuously indicates the radiation dose rate in the area.
b) A radiation monitoring device that continuously integrates the dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.
c) An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Radiation Protection Supervisor in the RWP.

Contrary to the above, on September 28, 2018, the licensee failed to implement Section 4.3.3(4)(a) of MRP04 and to implement TS 5.7.1. Specifically, the licensee failed to implement RWP 183019, as two individuals entered a HRA in the steam tunnel without being briefed of radiological conditions. Additionally, the licensee failed to make the individuals knowledgeable of the dose rate levels in the area when entering an HRA with dose rates >100 mrem/hr but <1000 mrem/hr as required by TS 5.7.1.b. Additionally, the conditions of TS 5.7.1.a or 5.7.1.c were not met.

Disposition: This violation is being treated as a Non-Cited Violation, consistent with Section 2.3.2 of the Enforcement Policy.

Failure to Establish Radiological Conditions for a Locked High Radiation Area Entry

Cornerstone	Significance	Cross–Cutting Aspect	Report Section
Occupational Radiation Safety	Green NCV 05000341/2018004–05 Open and Closed	[H.10] – Human Performance — Bases For Decisions	71124.01—Radiological Hazard Assessment and Exposure Controls

A self-revealed finding of very low safety significance and an associated NCV of TS 5.7.2, “High Radiation Area,” was identified when RP staff allowed workers to enter an area of a Locked High Radiation Area (LHRA) where radiological conditions had not been established.

<u>Description</u>:

On September 22, 2018, the licensee shutdown the plant for a refueling outage. On September 26, 2018, two workers were briefed by RP staff to enter the Reactor Water Cleanup (RWCU) heat exchanger room, which was a LHRA, to install drain lines on two valves and to drain a portion of the RWCU system. The individuals were briefed to an expected dose rate alarm of greater than (>) 1300 mrem/hour due to piping in the work area that they might encounter. The workers were instructed to leave the area if they received more than one dose rate alarm or if the initial alarm failed to clear after moving to a different location. The workers were not able to perform the planned work due to the receipt of multiple dose rate alarms. The workers exited the area and reported the issues encountered to RP staff. The RP staff performed follow-up surveys of the area, which measured general area dose rates of up to 860 mrem/hour. The survey also identified a RWCU strainer in the general work area that was 150 rem/hour on contact and 37 rem/hour at 30 centimeter (cm), and a pipe which measured 2.5 rem/hour on contact and 1.2 rem/hour at 30 cm.

The inspector noted that the surveys performed in the RWCU heat exchanger room after plant shutdown did not include dose rates in this work area. A discussion with the RP staff that performed the radiological briefing revealed that surveys taken prior to plant shutdown were used to brief the workers for the evolution. These included surveys from October 2013, November 2015, and June 2017. The June 2017 survey documented general area dose rates of up to 500 mrem/hour with the highest on-contact dose rates of 1200 mrem/hour and 30 cm dose rates of 600 mrem/hour. The briefing the workers received from the RP staff indicated that expected dose rates ranged from 60 mrem/hour to 1200 mrem/hour.

The RWCU system is designed to filter out radioactive material from the reactor coolant system and dose rates in areas near the RWCU system can change based on plant conditions. For example, during shutdown, the site experienced a crud burst that released radioactive material into the system that could reasonably be expected to affect dose rates in the RWCU system. The inspectors determined that the use of radiological information taken prior to plant shutdown did not ensure that current radiological conditions were established.

The inspectors determined that this issue was self–revealed because the follow-up surveys of the work area were performed in response to dose rate alarms received by the workers.

Corrective Actions: Immediate corrective actions included placing temporary administrative controls on HRAs susceptible to changing radiological changes to ensure appropriate surveys were conducted prior to allowing workers to enter these areas. The licensee also performed a human performance review of the issue and held a discussion of the issue at a site stand down. Planned corrective actions included a review of plant processes, and revisions as necessary, to ensure that changes in plant conditions are considered when determining when to perform radiological surveys to ensure compliance with TS requirements.

Corrective Action Reference: CARD 18–27799

Performance Assessment:

Performance Deficiency: The failure of RP staff to establish the radiological conditions in a LHRA before allowing workers to enter this LHRA was a performance deficiency.

Screening: The inspectors determined that the performance deficiency was more than minor because it was associated with the Program and Process attribute of the Occupational Radiation Safety cornerstone and adversely affected the cornerstone objective of ensuring the adequate protection of worker health and safety from exposure to radiation. Specifically, worker entry into areas where radiological conditions have not been established could lead to unintended dose.

Significance: The inspectors assessed the significance of the finding using IMC 0609 Appendix C, "Occupational Radiation Safety Cornerstone." The inspectors determined that the finding was of very low safety significance (i.e., Green) because: (1) it did not involve as-low-as-reasonably-achievable planning or work controls; (2) there was no overexposure; (3) there was no substantial potential for an overexposure; and (4) the ability to assess dose was not compromised.

Cross-Cutting Aspect: The finding had a cross-cutting aspect in the Bases for Decisions component of the Human Performance cross-cutting area, which states that leaders ensure that the bases for operational and organizational decisions are communicated in a timely manner. Specifically, for this finding, leaders did not communicate expected outcomes and potential problems with changes in radiological conditions on the RWCU system to RP technicians that perform the radiological surveys and conduct HRA briefings. [H.10]

Enforcement:

Violation: Technical Specification 5.7.2, "High Radiation Area," establishes controls for HRAs with areas accessible to individuals with radiation levels such that an individual could receive in 1 hour a dose equivalent >1000 mrem but < 500 rads at one meter from the sources of radioactivity. Technical Specification 5.7.2 requires, in part, that the requirements of TS 5.7.1 be met for these areas.

Technical Specification 5.7.1, "High Radiation Area," requires in part, that individuals entering HRAs be provided with or accompanied by either:

a) A radiation monitoring device that continuously indicates the radiation dose rate in the area.
b) A radiation monitoring device that continuously integrates the dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.
c) An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities

within the area and shall perform periodic radiation surveillance at the frequency specified by the Radiation Protection Supervisor in the RWP. Contrary to the above, on September 26, 2018, the licensee failed to establish dose rate levels and to make personnel knowledgeable of them prior to allowing entry into a HRA in accordance with TS 5.7.1.b. Specifically, individuals entered the RWCU heat exchanger room, which was a HRA, with radiation levels meeting the criteria of TS 5.7.2, with a radiation monitoring device that continuously integrated the dose rate in the area and alarmed when a preset integrated dose is received; however, dose rate levels in the area had not been established and the individuals had not been made knowledgeable of them. Additionally, the conditions of TS 5.7.1.a or 5.7.1.c were not met. Disposition: This violation is being treated as a Non-Cited Violation, consistent with Section 2.3.2 of the Enforcement Policy.

<u>71152—Problem Identification and Resolution</u>

Failure to Verify the Adequacy of the Design of the 4160 Volt Safety Bus 64B, 64C, 65E, and 65F Degraded Voltage Protection Schemes			
Cornerstone	Significance	Cross–Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000341/2018004–06 Open and Closed	None	71152—Problem Identification and Resolution
A finding of very low safety significance and associated NCVs of 10 CFR 50, Appendix B, Criterion III, "Design Control"; Technical Specification (TS) 3.3.8.1, "Loss of Power (LOP) Instrumentation"; TS 3.5.1, "ECCS [Emergency Core Cooling System] – Operating"; and TS 3.8.1, "AC [Alternating Current] Sources – Operating," were self-revealed when licensee staff failed to verify the adequacy of the design of the 4160 volt safety-related bus 64B, 64C, 65E, and 65F under voltage (degraded voltage) protection relaying circuitry to ensure the associated relaying did not operate during emergency diesel generator load sequencer operation. Specifically, during various loss of offsite power (LOP) and loss of coolant accident (LOCA) scenarios, a load shed signal from the degraded voltage protection relaying was identified to have prevented the associated residual heat removal (RHR) pump from automatically starting while the load sequencer was operating.			
<u>Description</u>: On October 18, 2018, the licensee performed emergency diesel generator 11 LOP/LOCA surveillance testing using a new revision to the surveillance testing procedure that initiated the test with the emergency diesel generator in a running condition. Previous revisions of the surveillance testing procedure initiated the test with the emergency diesel generator in a standby condition. The initial condition of the emergency diesel generator, whether running or in standby, was not expected to have any impact on the successful performance of the testing since design requirements required both initial conditions to be successful and the system design was believed to ensure successful testing results for both initial testing conditions. During the test, RHR pump 'A' immediately tripped while sequencing onto its associated 4160 volt safety bus (Bus 64B). Subsequently, on October 18, 2018, during performance of similar emergency diesel generator 12 LOP/LOCA surveillance testing, RHR pump 'C' also			

immediately tripped while sequencing onto its associated 4160 volt safety bus (Bus 64C). Both of these surveillance tests were conducted during a planned refueling outage to simulate a LOP signal in conjunction with a simulated LOCA signal to ensure the associated emergency diesel generator would automatically start and sequence on various safety-related equipment loads to the associated 4160 volt safety bus, including the RHR pumps.

Licensee troubleshooting identified a legacy design issue associated with the degraded voltage protection scheme. The issue was reported to the NRC in Event Notification (EN) 53674 as an unanalyzed condition on October 19, 2018.

The Fermi 4160 volt safety bus LOP instrumentation and associated load shedding and sequencing scheme was improperly configured such that under certain degraded voltage scenarios with a LOCA signal present, the degraded grid relays would actuate and the RHR pumps would automatically trip while being sequenced onto their associated 4160 volt safety bus (64B and 64C for Division 1; 65E and 65F for Division 2) and therefore would not automatically start and inject water into the reactor vessel as designed. This condition applied to all four 4160 volt safety buses' LOP instrumentation, all four emergency diesel generators, and all four RHR pumps, and the problem existed in Modes 1, 2, and 3. The RHR pumps could have been subsequently restarted in the main control room using control board hand-switches. Per Fermi's Updated Final Safety Analysis Report (UFSAR), degraded grid relays were not designed to operate during load sequencer operation.

The licensee performed a root cause evaluation and determined that the inadequate degraded voltage relay scheme was an unrecognized original design defect that existed since the plant was licensed in 1985. As a result, the emergency diesel generator degraded voltage load shed relay would have tripped the RHR pumps, and would have prevented the automatic initiation of low pressure coolant injection (LPCI) in the following scenarios:

- LOP followed by a LOCA after the emergency diesel generator attained rated speed and voltage;
- Degraded voltage followed by a LOCA after the emergency diesel generator attained rated speed and voltage; and
- LOCA followed by a LOP after the emergency diesel generator attained rate speed and voltage.

The licensee determined that the load sequencing of all other safety-related loads was unaffected. The licensee also determined that there were numerous previous opportunities to identify this issue. Most notably, in 2010 the 4160 volt safety bus degraded voltage circuits were modified to add a shorter degraded voltage time delay relay specific to LOCA conditions to improve the LPCI response time during a degraded voltage condition.

A number of TS LCOs were identified to have not been entered as required. These TSs included the following:

- TS 3.3.8.1, "Loss of Power (LOP) Instrumentation," requires that LOP instrumentation be operable for an under voltage condition with a LOCA while in Modes 1, 2, and 3. As a consequence of this design control issue, LOP instrumentation was not operable for an Undervoltage condition with a LOCA in Modes 1, 2, and 3 and the associated LCO actions were not completed within the prescribed allowed outage time (AOT).

- TS 3.5.1, "ECCS – Operating," requires that each ECCS injection/spray subsystem shall be operable in Modes 1, 2, and 3. As a consequence of this design control issue, two or more low pressure ECCS injection/spray subsystems were not operable in Modes 1, 2, and 3 and the associated TS LCO actions were not completed within the prescribed AOT.

- TS 3.8.1, "AC Sources – Operating," requires that two emergency diesel generators per division shall be operable in Modes 1, 2, and 3. As a consequence of this design control issue, all four emergency diesel generators were not inoperable in Modes 1, 2, and 3 and the associated TS LCO actions were not completed within the prescribed AOT.

Corrective Actions: A physical modification was installed on all four 4160 volt safety buses to block the associated degraded voltage load shedding scheme when the 4160 volt safety bus offsite power supply breaker was open.

Corrective Action Reference: CARD 18–28990

Performance Assessment:

Performance Deficiency: The failure to verify the adequacy of the design of 4160 volt safety-related bus 64B, 64C, 65E, and 65F under voltage (degraded voltage) protective relaying circuitry to ensure the associated relaying did not operate during emergency diesel generator load sequencer operation was a performance deficiency.

Screening: The inspectors determined that the performance deficiency was more than minor because it was associated with the Design Control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, the licensee failed to recognize a latent design flaw in the 4160 volt safety bus degraded voltage protection circuitry that rendered the LOP instrumentation for degraded voltage, all four emergency diesel generators, and all four RHR pumps inoperable since they were not fully capable of performing their intended safety function since original plant construction.

Significance: The inspectors assessed the significance of the finding using IMC 0609, Appendix A, "Significance Determination Process (SDP) for Findings at Power," Exhibit 2 for the Mitigating Systems Cornerstone. A detailed risk evaluation was required based on the deficiency affecting the design of a mitigating structure, system, or component that was rendered inoperable, a loss of system and/or function, and an actual loss of function of at least a single train for greater than its TS AOT or two separate safety systems out-of-service for greater than their TS AOTs.

A Senior Reactor Analyst (SRA) completed a detailed risk evaluation using the NRC Standardized Plant Analysis Risk (SPAR) model for Fermi, version 8.52. The SRA modeled the degraded plant condition as a failure of all four RHR pumps to start following a large break LOCA initiating event. The LPCI function of the RHR pumps was assumed to be able to be recovered in the control room by manually starting one or more RHR pumps. The SRA assumed this would be a highly reliable action for all other LOCA events and that the change in core damage frequency for these scenarios would be negligible.

The change in core damage frequency was calculated to be less than 1E–7/year. The dominant core damage sequence was a large break LOCA followed by a failure of all LPCI pumps due to the performance deficiency and the random failure of the low pressure core spray system. The results were conservative because the concurrent condition of a LOP or degraded voltage event would also need to occur to result in the failure of the RHR pumps, but was not modeled. Cross-Cutting Aspect: No cross-cutting aspect was assigned to this finding because the finding did not reflect current licensee performance.
<u>Enforcement</u>: Violation: Title 10 CFR 50, Appendix B, Criterion III, "Design Control," requires, in part, that the licensee 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. Fermi UFSAR Section 8.2.2.5.3, "Response to Degraded Grid Condition," states, in part, that degraded grid relaying is not designed to operate during emergency diesel generator load sequencer operation. Contrary to the above, from original plant construction until October 23, 2018 (Division 1) and October 24, 2018 (Division 2), the licensee failed to verify the adequacy of design of the 4160 volt safety-related bus 64B, 64C, 65E, and 65F under voltage (degraded voltage) protective relaying circuitry to ensure the associated degraded grid relaying did not operate during emergency diesel generator load sequencer operation. Specifically, during various LOP and LOCA scenarios, a load shed signal from the degraded voltage protection relaying would have prevented the associated RHR pump from automatically starting while the load sequencer was operating as required. In addition, LCOs associated with TS 3.3.8.1, "Loss of Power (LOP) Instrumentation;" TS 3.5.1, "ECCS – Operating;" and TS 3.8.1, "AC Sources – Operating," were not completed within the prescribed AOT. Disposition: This violation is being treated as a Non-Cited Violation, consistent with Section 2.3.2 of the Enforcement Policy.

Observation – Adverse Trend in Operability Determination Program Implementation	71152—Semiannual Trend Review
The inspectors identified an adverse trend in operability determination program implementation following a review of operability and functionality assessments documented in the corrective action program over the 6 month period of June 2018 to December 2018. The Fermi TSs defined operability in the following manner: "a system, subsystem, division, component, or device shall be operable or have operability when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s)." Similar to operability, functionality referred to structures, systems, or components not listed in the TSs.	

The operability determination process, as specified in procedure MQA11–100, "Operability Determination Program," is intended to continuously assess the operability and functionality of structures, systems, or components (SSCs). Whenever a SSC's operability or functionality is in question, an immediate assessment is to be made by a licensed senior reactor operator (SRO). Depending on the circumstances, a past operability or functionality assessment is then requested to determine how long that condition existed. If more information is needed to confirm or deny an immediate operability or functionality assessment, an engineering functional analysis (prompt operability assessment) is to be requested.

Whenever a SSC deficiency is identified, a condition assessment resolution document (CARD) is required by procedure MQA11, "Corrective Action Program," to be written to document the deficiency. An SRO (generally the shift manager) then performs and documents an operability and/or functionality assessment to determine if a reasonable expectation of operability and/or functionality exists given the identified deficiency.

The inspectors identified multiple examples of documented operability determinations that did not justify a reasonable expectation of operability and/or functionality for the SSC in question. Examples included:

- On May 25, 2018, CARD 18–24184 documented a procedure deficiency associated with the visual examinations of snubbers. In particular, the CARD identified that snubber T46–3093–G11 associated with the safety-related standby gas treatment system was listed to only impact one division when removed for maintenance, when it actually impacted both divisions as it was installed on common piping. The inspectors noted the CARD was not annotated to require a past operability evaluation, but was only evaluated as an administrative issue. The licensee performed a subsequent past operability evaluation that determined the snubber was previously removed from service for maintenance when neither division of standby gas treatment was required to be operable. Although there were no previous actual impacts, the outcome was not known until questioned by the inspectors. CARD 18–25018 was also written to determine the extent of condition of the issue, which identified other snubbers associated with both divisions of standby gas treatment that were identified to only impact one division. A past operability evaluation was performed on those snubbers and no adverse impacts were identified.
- On June 1, 2018, CARD 18–24354 documented a small lube oil leak from an emergency diesel generator 11 lube oil filter piping flexible coupling. The shift manager's review of the CARD documented that the lube oil leak was small while in a standby condition and therefore did not challenge the operability of the emergency diesel generator. The review did not include an assessment of the lube oil leak during running conditions. A pencil stream lube oil leak developed during the next emergency diesel generator 11 surveillance test and the system was shut down and declared inoperable. A self-revealed finding and associated NCV (NCV 05000341/2018002–03) was documented in an NRC integrated inspection report (ADAMS Accession Number ML18229A293).
- On June 1, 2018, CARD 18–24353 documented a small lube oil leak from a flange connection downstream of the standby lube oil pump on emergency diesel generator 12. Similar to CARD 18–24354 discussed above, the operability assessment did not include the potential impact on the emergency diesel generator during a running condition. Additional licensee review determined there was no impact based on the

location in the system being isolated by a check valve while the emergency diesel generator was running.

- On June 7, 2018, CARD 18–24509 documented several cracked electrolyte covers on the Division 2 130/260 volt safety-related batteries. The CARD was annotated as license based “No”, which meant no operability determination was needed and none was performed. A subsequent operability determination was performed on June 18, 2018, when questioned by the inspectors, and which determined there were no adverse impacts and that the batteries were capable of performing their safety function.
- On July 14, 2018, CARD 18–25397 documented a trip of peaker fuel oil forwarding pump 1 and that peaker fuel oil forwarding pump 2 did not automatically start as designed, which involved the fuel supply for combustion turbine generator 11–1 credited for station blackout conditions. The CARD was annotated as license based “No”, which meant no operability determination was needed and none was performed. A subsequent operability determination was performed on July 18, 2018, when questioned by the inspectors, and which determined there were no adverse impacts as the direct current emergency fuel forwarding pump remained available to supply combustion turbine generator 11–1.
- On September 13, 2018, CARD 18–00146 documented a high water level alarm and oil detected light on the sump pump control panel for underground cable vault (manhole) 16497. An operability determination was performed that indicated there were no adverse impacts because the cable vault itself was new and the medium voltage cables running through it had been recently replaced. The inspectors noted the cable vault was an older vault and that it did not contain medium voltage cables. The licensee determined the shift manager misunderstood which vault was listed in the CARD, its associated condition, the type of cabling, and did not validate assumptions using available reference material prior to making an immediate operability determination. The operability evaluation was re-performed on September 15, 2018, using updated information and determined the low voltage cables were not impacted.
- On October 26, 2018, CARD 18–28795 documented an estimated 200 drop per minute steam leak from the packing region of high pressure coolant injection (HPCI) outboard steam admission valve E4150–F003. The licensee performed an immediate operability determination that concluded the leakage was within the total allowable HPCI system leakage rate, and therefore the system remained operable. The inspectors noted that the immediate operability determination was based on the system pressure at the time of discovery, which was estimated to be about 50 pounds per square inch gauge (psig) less than the nominal system pressure of 1025 psig. The operability determination did not discuss the potential leak rate changes given a 50 psig increase in pressure and whether the HPCI system would remain within allowable total system leakage limits. The inspectors also noted a monitoring plan had not been established to evaluate changes to the leak rate to ensure continued operability given the degraded condition. The licensee provided additional engineering judgement that the HPCI system would remain operable given a pressure increase and developed a monitoring plan to ensure the basis for continued operability remained valid.

Additional examples identified by the licensee and documented in the corrective action program, included, but were not limited to:

- On May 15, 2018, CARD 18–23890 documented a main turbine steam chest temperature instrument that was not working properly. The functionality assessment documented in the CARD, however, addressed an issue associated with drywell bulk temperature, which was unrelated to the main turbine steam chest temperature instrument. The licensee documented in CARD 18–24053 that an incorrect assessment was performed because the shift manager had two CARDs open simultaneously when performing operability and functionality assessments and documented the assessment in the wrong location for the wrong component.

The licensee entered the NRC–identified adverse operability determination trend into their corrective action program as CARD 18–25732. An evaluation was performed that included interviews with shift managers and various organizations independently reviewed a sample of CARDs for several months to assess the quality of operability determinations and functionality assessments. The cause was determined to be a lack of attention to detail. Corrective actions included, but were not limited to, reinforcement of site standards and expectations and periodic checks by operations management, licensing, and quality assurance staff.

The inspectors concluded that these examples were minor procedural non-compliances that were not subject to NRC enforcement action.

<u>71153—Follow–Up of Events and Notices of Enforcement Discretion</u>

Rainwater Intrusion into Degraded Metal-Clad Enclosure for Bus 1–2B Results in Partial Loss of Offsite Power, Reactor Scram, and Emergency Core Cooling System Initiation			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Green FIN 05000341/2018004–07 Open and Closed	[H.12] – Human Performance – Avoid Complacency	71153—Follow–Up of Events and Notices of Enforcement Discretion
A finding of very low safety significance was self-revealed on April 14, 2018, when rainwater intrusion into a degraded metal-clad enclosure for the Bus 1–2B switchgear in the Division 1 120 kilovolt (kV) switchyard caused a partial loss of offsite power, reactor scram, and emergency core cooling system initiation. Specifically, the licensee failed to adequately maintain the physical integrity of the metal-clad enclosure for Bus 1–2B and did not recognize the vulnerability to plant operation.			

<u>Description</u>:

On April 14, 2018, while at full power with heavy rain and wind in the area, a lockout of station service transformer 64 occurred that resulted in a partial loss of feedwater and an automatic reactor scram. As designed, the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems both initiated and briefly injected into the reactor vessel to restore reactor water level. All control rods fully inserted and plant systems responded as designed given the initiating event. The lockout of station service transformer 64 also caused Division 1 of the 4160 volt safety buses (64B and 64C) to lose power. Emergency diesel generators 11 and 12 automatically started and provided power to the Division 1 safety buses. Offsite power to the redundant Division 2 safety buses was not affected. The cause of the event was determined to be water intrusion into a breaker in the Division 1 120 kV switchyard that resulted in the lockout of station service transformer 64. The licensee reported the event in Event Notification (EN) 53336 and Licensee Event Report (LER) 2018–002.

Division 1 offsite electrical power is normally supplied by three offsite power lines that are stepped down through transformer 1 (120 kV to 13.2 kV) to Bus 11. Power is then routed through the normally closed 'D' breaker of Bus 11 to station service transformer 64 (13.2 kV to 4160 volt). Station service transformer 64 supplies power to various 4160 volt buses including Division 1 4160 volt safety-related buses 64B and 64C. Division 2 offsite electrical power is supplied through an independent and physically separate 345 kV switchyard. In the event of a station blackout, one or more permanently installed combustion turbine generator peaker units are available to provide power from 13.8 kV peaker Bus 1–2B (through the normally open A6 breaker) to station service transformer 64 and to the Division 1 4160 volt buses.

The licensee performed a root cause evaluation and determined that the organization failed to recognize the degradation of the 13.8 kV switchgear metal-clad Bus 1–2B enclosure as a potential failure mechanism of the A6 switchgear. Rain water intrusion through the metal-clad enclosure of peaker Bus 1–2B wetted the normally open A6 breaker, which caused a phase to ground fault that was sensed by the normally closed 'D' breaker on Bus 11. The 'D' breaker subsequently opened and caused the subsequent loss of station service transformer 64 and plant transient.

An identified contributing cause was that operations, maintenance, and engineering management tolerated low standards for the material condition of 13.8 kV outdoor metal-clad switchgear. Other contributing causes included: 1) a design flaw with the bus duct entry being near the roof ridge vent, which in conjunction with the design of the roof ridge vent, resulted in water flowing inside the cubicle joint roof cap; 2) the failure of Detroit Edison using available internal and external operating experience to identify the outdoor metal-clad switchgear enclosure as a failure mechanism; and 3) Distribution operations procedures and specifications for switchyard inspections and maintenance utilized on Fermi 2 equipment that were not included in Fermi 2 design and procedure controls.

Procedural guidance was available from Detroit Edison distribution operations that specifically addressed the inspection of metal-clad switchgear enclosures on a monthly basis and the performance of maintenance, as necessary, based on internal Detroit Edison operating experience. Additionally, similar rainwater intrusion events in outdoor switchgear enclosures that resulted in the loss of equipment and/or plant transients had occurred at other nuclear plants in the industry. Moisture intrusion was previously identified in the A6 cubicle in 2002 and the roof was sealed with an epoxy coating. However, no preventative maintenance task was created to ensure the roof coating was maintained. Onsite programs such as the preventative maintenance, aging management, and Maintenance Rule program did not consider the potential impacts of water intrusion into Bus 1–2B switchgear. Also, procedure 27.000.05, "Operator Rounds," contained guidance for the identification of equipment deficiencies, including water leaks in the vicinity of equipment that could be adversely affected by moisture.

Corrective Actions: Metal-clad bus enclosure 1–2B was temporarily repaired and the enclosure roof was subsequently replaced to prevent rainwater intrusion. The switchgear enclosure was also added to the site's preventative maintenance, aging management, and Maintenance Rule programs for long-term monitoring and maintenance in coordination with

Detroit Edison distribution operations. Extent of condition walkdowns of similar enclosures were also performed. Corrective Action Reference: CARD 18–23026
Performance Assessment: Performance Deficiency: The failure to adequately maintain and recognize the vulnerability of the metal-clad enclosure for Bus 1–2B switchgear in accordance with site standards and available procedural guidance was a performance deficiency. Specifically, rainwater intrusion through the roof enclosure of Bus 1–2B in the 120 kilovolt switchyard resulted in a loss of station service transformer 64, a reactor scram, and ECCS initiation. Screening: The inspectors determined that the performance deficiency was more than minor because it was associated with the Equipment Performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the failure to adequately maintain the metal-clad enclosure for Bus 1–2B resulted in rainwater intrusion that caused a partial loss of offsite power, reactor scram, and ECCS initiation. Significance: The inspectors assessed the significance of the finding using IMC 0609, Appendix A, "The Significance Determination Process (SDP) For Findings At–Power," and answered "No" to the transient initiator question of Exhibit 1, "Initiating Events Screening Questions." Therefore, the inspectors determined that the finding was of very low safety significance (i.e., Green). Cross-Cutting Aspect: The finding had a cross-cutting aspect in the Avoid Complacency component of the Human Performance cross-cutting area, which states that the licensee will recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes and that individuals implement appropriate error reduction tools. Specifically, for this finding, operations, maintenance, and engineering personnel tolerated low standards for the material condition of the 13.8 kV outdoor metal-clad switchgear, despite routine walkdowns and inspections of the Division 1 switchyard, and did not recognize the inherent risk of a water intrusion event. [H.12]
Enforcement: The inspectors did not identify a violation of regulatory requirements associated with this finding.

EXIT MEETINGS AND DEBRIEFS

The inspectors confirmed that proprietary information was controlled to protect from public disclosure. No proprietary information was documented in this report.

- On October 5, 2018, the inspectors presented the radiation protection program inspection results to Mr. M. Caragher, Fermi Plant Manager, and other members of the licensee staff.

- On October 16, 2018, the inspectors presented the in-service inspection activities results to Mr. K. Mann, Supervisor, Nuclear Compliance, and other members of the licensee staff.

- On November 29, 2018, the inspectors presented the emergency preparedness program inspection results to Mr. N. Avrakatos, Radiological Emergency Response Preparedness Manager.

- On December 18, 2018, the inspectors discussed the completed 2018 Licensed Operator Requalification Program annual operating test inspection results with Mr. M. Donigian, Operations Training Program Supervisor.

- On January 14, 2019, the inspectors presented the quarterly integrated inspection results to Mr. K. Polson, Senior Vice President and Chief Nuclear Officer, and other members of the licensee staff.

DOCUMENTS REVIEWED

71111.01—Adverse Weather Protection

- CARD 18-29342; Failure of the South Boiler to Restart; 11/16/2018
- Procedure 20.000.01; Acts of Nature; Revision 54
- Procedure 27.000.04; Freeze Protection Lineup Verification; Revision 53
- Procedure 27.000.07; Cold Weather Operations; Revision 6

71111.04—Equipment Alignment

- Drawing 6M721-5444; Emergency Equipment Cooling Water Division 1; Revision CC
- Drawing 6M721-5729-1; Emergency Equipment Cooling Water Division 1; Revision BL
- Procedure 23.127; Reactor Building Closed Cooling Water / Emergency Equipment Cooling Water System; Revision 145
- Procedure 23.129; Station and Control Air System; Revision 115
- Procedure 23.203; Core Spray System; Revision 61

71111.05AQ—Fire Protection Annual/Quarterly

- Procedure FP-RHR-1-50; RHR Complex, Division 1 Pump Room, Zone 50, Elevation 590’0”; Revision 5
- Procedure FP-RHR-1-12-EDG; RHR Complex, EDG 12 Room, Elevation 590’0”; Revision 7
- Procedure FP-RHR-1-12-OS; RHR Complex, EDG 12 Oil Storage Room, Elevation 590’0”; Revision 4
- Drawing 6A721-2403; Fire Protection Evaluation Reactor and Auxiliary Buildings First Floor Plan, Elevation 583’6”; Revision T
- Procedure FP-RB-Drywell; Drywell; Revision 3
- MOP11; Fire Protection; Revision 22
- Procedure FP-TB; Turbine Building; Revision 10
- CARD 18-30190; Florescent Lamp Swinging, Contacting Overhead Pipe; 12/18/2018
- MOP11-100; Fire Protection Implementation; Revision 5

71111.06—Flood Protection Measures

- CARD 17-00666; Sump Pump Panel Y4100P033 Auto Non-Functional; 5/21/2017
- CARD 18-00146; Oil Detected Light on for Y4100P033; 09/13/2018
- CARD 18-00202; Manhole 16947 High Level Alarm Lit with Oil Detected Light; 7/27/2018
- CARD 18-00203; Manhole 16947A Sump Pump Feed Breaker Trip; 7/27/2018

- CARD 18-25833; Confirmed Water Standing in Cable Vault 16947A; 8/2/2018
- CARD 18-25863; Partial Complete WO 47169678 Cable Vault Inspections; 8/3/2018
- Fermi 2 Control Room Log; 7/19/18
- WO 49406005; License Renewal – Perform Periodic Inspections of Manhole 16946A/B/C for Water Accumulation; 12/13/2017
- Drawing M-2218; Sub-Basement Plan Elevation 540’0”
- CARD 18-30216; NRC Identified Gap in Seal on Watertight Door T1-32; 12/18/2018
- CARD 17-25515; Drains Backed Up Causing Overflow on to Floor; 08/22/2017
- CARD 18-24957; Floor Drains Overflowed From Pumping Down Sump DO-74; 06/26/2018
- Human Performance Review Board Minutes; 06/27/2018
- CARD 18-24957-01; Complete MRFF Review; 06/28/2018
- Drawing RM-583-RX-AX; Reactor – Auxiliary Building First Floor; 07/21/1982
- DC-5110; Main Steam and Feedwater Line Break in the Steam Tunnel; Revision A

<u>71111.07—Heat Sink Performance</u>

- CARD 18-27804; NQA-RF19 Arc Strike Identified During Weld Inspection; 10/04/2018
- CARD 18-30289; Damage to South RHR Heat Exchanger Room Platform Beam B88 at Top of Steel (TOS) Elevation 613’5”; 12/20/2018
- WO 25976827; Eddy Current Testing of Heat Exchanger; 09/29/2018

<u>71111.08—In-Service Inspection Activities</u>

- CARD 17-23237; Foreign Material in Region between Shroud Support Plate and Jet Pump Tail Pipe; 04/08/2017
- CARD 18-27269; #13EDGSW Pump Column Flanges Have Some Funnel Shaped Mass Around Bolting; 09/25/2018
- CARD 18-27271; On #13 EDGSW Pump Min Flow Line Approximately Three Inches Broke Off; 09/25/2018
- CARD 18-27590; Indication Noted in Core Differential Pressure/Standby Liquid Control Safe End Weld 5-315
- Procedure 39.NDE.001 Revision 29A; Liquid Penetrant Examination, Solvent Removable; 05/01/2018.
- Procedure 39.NDE.002 Revision 25A; Magnetic Particle Examination by the AC/DC Yoke Method; 05/01/2018.
- Procedure GEH-PDI-UT-1, Revision 12; PDI Generic Procedure for the Ultrasonic Testing of Ferritic Pipe Welds; 07/19/2018
- Procedure GEH-PDI-UT-2, Revision 12; PDI Generic Procedure for the Ultrasonic Testing of Austenitic Pipe Welds; 07/19/2018
- Welding Procedure Specification (WPS) A11-3.1, Rev 2 GTAW/SMAW P-No 1; 10/02/2012
- WO 34306179 Replacement of EDG 14 SW Outlet Valve Component ID R3000F139D; 04/08/2017
- WO 44879308; Pipe Replacement of Division 2 EECW HX Service Water Return Piping Component ID MK-P45-3353-10; 04/04/2017
- WO 47414895; Reportable Magnetic Particle Indication Found on Support Lug SW-PS-2-A2-AA4; 04/10/2017

<u>71111.11—Licensed Operator Requalification Program and Licensed Operator Performance</u>

- Fermi 2 Evaluation Scenario SS-OP-904-1812; Main Steam Line Flow Transmitter Fail; Revision 0

- Fermi 2 Evaluation Scenario SS-OP-904-1822; Earthquake/Loss of 72B/S. H2 Seal Oil Pump Trip/Steam Leak in Reactor Building Steam Tunnel/MSIVs Fail to ISO/HPCI Steam Leak – No Iso/2 Areas MSO/1 SRV Fail; Revision 0
- Procedure 54.000.07; Core Performance Parameter Check; Revision 59

<u>71111.12—Maintenance Effectiveness</u>

- CARD 18-28371; IPCS Point G33DF 1055 Bad; 10/16/2018
- CARD 18-26442; Routine Oil Samples for HPCI and RCIC Scheduled at the Same Time; 08/26/2018
- CARD 17-27658; NQA – HPCI Turbine Exhaust High Pressure Switches Out of APT During the HPCI SSO; 09/14/2017
- CARD 17-26996; E41N017A As Found Data Out of APT; 08/21/2017
- CARD 17-26997; As Found Data for E41N017B was Found Out of APT; 08/21/2017
- CARD 18-23026; Reactor Scram Due to Loss of 64 Transformer; 04/14/2018
- Letter NRC-17-0070; Non-Functional Mechanical Draft Cooling Tower Fan Brakes Leads to HPCI Being Declared Inoperable and Loss of Safety Function which is a Condition Prohibited by Technical Specifications and Loss of Safety Function; 11/03/2017
- Fermi 2 Business Practice FBP-68; System Health Program; Revision 21

<u>71111.13—Maintenance Risk Assessments and Emergent Work Control</u>

- CARD 18-27806; Torus Vapor Space; 10/04/2018
- CARD 18-28240; Light Surface Rust Noted Throughout the Interior of the Ring Header; 10/13/2018
- CARD 18-29568; New Replacement Valve Stem and Disc Dropped from Rigging with Personnel Injury; 11/29/2018
- CARD 18-29569; Damaged Equipment Resulting From Dropped E1150F068B Stem; 11/29/2018
- CARD 18-29570; Snubber E11-3151-G17 Damaged from E1150F068B Stem Drop; 11/29/2018
- CARD 18-29580; Damaged Grating Resulting from Dropped E1150F068B Stem; 11/29/2018
- CARD 18-29583; Damaged Support Beam Resulting from Dropped E1150F068B Stem; 11/29/2018
- CARD 18-29583; Damaged Support Beam Resulting from Dropped E1150F068B Stem; 11/29/2018
- CARD 18-29675; Main Generator Rotor Ground Fault Alarm; 12/02/2018
- CARD 18-29870; Reactor Building to Suppression Chamber Vacuum Breaker UFSAR Nonconformance; 12/08/2018
- CARD 18-30152; Minor Crack in Rotor Endwinding; 12/16/2018
- CARD 18-30330; Adverse Trend in Rigging and Lifting Practices; 12/21/2018
- CARD 18-30330; Adverse Trend in Rigging and Lifting Practices; 12/21/2018
- DCP 80069; Repair of Beam B81 in the South RHR Heat Exchanger Room; Revision 0
- Drawing 5M721-6223; 24” – 150# Powell Globe Valve with Limitorque Operator; Revision F
- Drawing 6C721-2510; Reactor Building RHR Heat Exchanger Platforms Unit – 2; Revision AA
- Drawing 6C721-2672; Reactor Building RHR Heat Exchanger Platforms Unit – 2; Revision E
- Drawing 6M621-5739-1; Nitrogen Inerting System; Revision AE
- EFA-E11-18-004; EFA for Division 2 RHR Heat Exchanger Damaged Support Structure Beams; Revision 0
- MDI-055; Fundamentals of Maintenance; 08/24/2018
- MGA24001; Pre-Job Brief Checklist; Revision 8

- MMA07; Hoisting, Rigging, and Load Handling; Revision 25
- MMA24; Maintenance Conduct Manual – Material Handling; Revision 4
- MWC15; Elevated Risk Management; Revision 17
- Procedure 23.425.01; Primary Containment Procedures; Revision 79
- Procedure 32.RIG.18; Guidelines and Practices for the Use of Hoisting and Rigging Equipment; Revision 15
- Procedure 32.RIG.18; Guidelines and Practices for the Use of Hoisting and Rigging Equipment; Revision 15
- Procedure 35.118.004; Main Unit Generator Disassembly and Reassembly; Revision 6
- Procedure 35.118.004; Main Unit Generator Disassembly and Reassembly; Revision 6
- Procedure ARP 4D13; Main turbine VIB High; Revision 22
- Underwater Engineering Services, Inc. Quality Assurance Records Index; Torus Underwater Desludge, IWE/Coatings inspection and Coating Repair
- WO 48693770; Extensive Coating Defects Identified During Torus Underwater Inspection; 09/18/2017
- WO 48693800; Inspect and Repair Torus Coating, Above Water; 09/18/2017
- WO 52341215; Remove TM 18-0030 (Restore E1150F068B Valve Internals); 11/07/2018
- WO 52621477; Condition / Assessment and Repair of Generator Field; 12/16/2018

71111.15—Operability Determinations and Functionality Assessments

- CARD 18-25151; Electromagnetic Compatibility Qualification Failure – 1E Isolators; 07/03/2018
- CARD 18-25563; Intermittent Indicated Jet Pump Loop Flow Mismatch; 07/21/2018
- CARD 18-25638; Adverse Impact Resulting From Partial Implementation of EDP-37536
- CARD 18-25642; Develop Procedure for Alternate Jet Pump Loop Flow Calculation; 07/24/2018
- CARD 18-26407; Did not Meet Acceptance Criteria in Step 6.6.14.4 in Surveillance 44.090.001; 08/24/2018
- CARD 18-26835; Momentary Loss of Indication on B31R613 Recirculation B Loop Flow Indicator; 09/11/2018
- CARD 18-27165; Vendor Calculated Negative Margin Close Stroke; 09/22/2018
- CARD 18-27188; MSIV Line B Exceeded its Repair Guideline Penetration X-7B; 09/23/2018
- CARD 18-27189; MSIV Line D Unable to Obtain Test Pressure Penetration x7D; 09/23/2018
- CARD 18-27477; Indications on Removed Stem; 09/28/2018
- CARD 18-27774; Packing Leak From E2150F005B; 10/04/2018
- CARD 18-27941; Stem on Division 2 INBD ISO MOV; 10/07/2018
- CARD 18-28038; Packing Leak Observed from Division 2 Core Spray Inboard Isolation Valve E2150F005B During LLRT; 10/09/2018
- CARD 18-28289; Upon Actuator Removal Discovered Water in the Compensator Assembly; 10/14/2018
- CARD 18-28405; E2150F005B Packing Leak; 10/17/2018
- CARD 18-28795; Steam Leak Identified on E4150F003; 10/26/2018
- CARD 18-29027; E1150F068B Did Not Close as Expected; 11/02/2018
- CARD 18-29043; E1150F068B Seat Ring Indications After Failed Valve Stem; 11/04/2018
- CARD 18-29057; 2018 LIR: Analytical Low Margin Inboard MSIVs; 11/05/2018
- EFA-E11-18-004; EFA for Division 2 RHR Heat Exchanger Damaged Support Structure Beams; Revision 0
- EFA-E21-18-003; E2150F005B Packing Leak in Modes 4 and 5; Revision 0
- MES27; Verification of System Operability; Revision 18
- Procedure 23.208; RHR Complex Service Water Systems; Revision 118

- TE-B21-18-024; Evaluation of the MSIVs N2 Pneumatic Supply Pressure; Revision 0
- TE-B21-18-053; Past Operability Evaluation for the Jet Pump Flow Instrumentation; Revision 0
- TE-E11-18-099; Past Operability Determination for E1150F068B, RHR Division 2 Heat Exchanger Service Water Isolation Valve for CARD 18-29027 "E1150F068B Did Not Close as Expected"; Revision 0
- WO 48233388; EDP-37536 Control Room Work at Panel H11P807 Remove Isolators; 07/18/2017
- CARD 18-28624; Water Dripping at Wall Penetration 152; 10/22/2018
- WO 52188792; Water Dripping at Wall Penetration 152; 10/22/2018
- CARD 18-28714; RHR Return Piping Seals Not Designed as Water Barriers; 10/25/2018
- Drawing 6M721-5706-3; RHR Service Water Make Up Decant and Overflow Systems; Revision AF
- Drawing 6M721-5706-1; Residual Heat Removal Division II; Revision AK

71111.18—Plant Modifications

- 24.307.01; EDG11 Loss of Offsite Power, Inserted Step to Close Undervoltage Cutout Switch Prior to Resetting UV Trip Targets to Correct Sequence; Revision 52
- 24.307.02; Procedure Change Based on Plant Mod and User Feedback; Revision 56
- 24.307.03; Added Note to Trigger Date Recorder and Clarified Initiating Event for Recorder; Revision 53
- 24.307.04; Added Note to Trigger Date Recorder and Clarified Initiating Event for Recorder; Revision 50
- 50.59 18-0247; EDP 80065/LCR 18-060-UFS; Revision 0
- 50.59 Evaluation; Temporary Modification (TM 18-0030) of Residual Heat Removal Division 2 Heat Exchanger Service Water Outlet Isolation Valve (E1150F068B); Revision 0
- 50.59 Evaluation; Temporary Modification of Residual Heat Removal Division 2 Heat Exchanger Service Water Outlet Isolation Valve E1150F068B; Revision 0
- CARD 18-28245; C RHR Pump Failed to Start During Surveillance Testing; 10/13/2018
- CARD 18-28457; RHR Pump 'C' Tripped After Auto Start During 24.307.02; 10/18/2018
- CARD 18-28495; RHR Pump 'A' Failed to Start as Expected During 24.307.02; 10/18/2018
- CARD 18-28572; Degraded Grid Relaying is Not Bypassed when the EDGs are Supplying their Respective ESF Bus; 10/20/2018
- DCP 80065; Permanent Plant Modification to Ensure that the 4160V ESF Bus Degraded Voltage Load Shed Logic is Inhibited when Offsite Power is Not Connected; Revision 0
- Design Change Package 18-0030; Removal of E1150F068B Valve Internals and Installation of Seal Welded Plate to Lower Bonnet Bushing Opening; Revision 0
- Design Change Package; Removal of E1150F068B Valve Internals and Installation of Seal Welded Plate to Lower Bonnet Bushing Opening; Revision 0
- Drawing 6I721-2111-01; Control Rod Drive Water Pump 'A' C1106C001A; Revision O
- Drawing 6I721-2201-01; Residual Heat Removal Pump 'A' E1102C002A; Revision T
- Drawing 6I721-2201-03; Residual Heat Removal Pump 'C'; Revision T
- Drawing 6I721-2205-02; RHR Relay Logic 'A' Circuit Part 1; Revision X
- Drawing 6I721-2205-03; RHR Relay Logic 'A' Circuit Part 2; Revision T
- Drawing 6I721-2211-01; Core Spray Pump 'A' E2101C001A; Revision O
- Drawing 6I721-2572-13; 4160V ESS Bus 64C-POS 'C8'; Revision 8
- Drawing 6I721-2572-14; 4160V ESS Bus 64C – POS 'C9"; Revision P
- Drawing 6I721-2572-15; 4160V ESS Bus '64C' – POS 'C6'; Revision U
- Drawing 6I721-2572-15; 4160V. ESS Bus "64C" – POS "C6"; Revision U
- Drawing 6I721-2572-28; 4160V ESS. Buses 64B and 64C Load Shedding Strings; Revision U
- Drawing 6I721-2572-56; 4160V ESS Bus 64B, POS 'B12'; Revision I

- Drawing 6I721-2572-57; 4160V ESS Bus 64B POS ‘B4’; Revision M
- Drawing 6I721-2578-07; Relaying and Metering Diagram 4160V ESS Bus 64C; Revision P
- Drawing 6SD721-2500-01; Plant 4160V and 480V System Service; Revision BS
- EDP-80065; Permanent Plant Modification to Endure that the 4160V ESF Bus Degraded Voltage Load Shed Logic is Inhibited when Offsite Power is Not Connected; Revision 0
- Procedure 20.300.SBO; Loss of Offsite and Onsite Power; Revision 27
- Procedure 23.208; RHR Complex Service Water Systems; Revision 122
- Procedure 24.307.01; Emergency Diesel Generator 11 – Loss of Offsite Power and ECCS Start with Loss of Offsite Power Test; Revisions 44, 49, and 50
- Procedure 24.307.04; Emergency Diesel Generator 14 – Loss of Offsite Power and ECCS Start with Loss of Offsite Power Test; Revision 50
- TE-R14-18-088; Past Operability Review of Degraded Grid Relaying and RHR Pump Motor Start; Revision 0
- WO 52174704; EDP-80065, Modification to Bus 64B Load Shedding Ckt; 10/21/2018
- WO 52174747; EDP-80065, Modification to Bus 64C Load Shedding Ckt; 10/21/2018
- WO 52174748; EDP-80065, Modification to Bus 64E Load Shedding Ckt; 10/21/2018
- WO 52174749; EDP-80065, Modification to Bus 64F Load Shedding Ckt; 10/21/2018

<u>71111.19—Post Maintenance Testing</u>

- CARD 14-22772; 3D76 Control Rod Over-Travel Received on Rod 30-15; 03/24/2014
- CARD 18-00087; DGSW Heat Exchanger Drain Line Clogged; 10/15/2018
- CARD 18-27165; Vendor Calculated Negative Margin Close Stroke; 09/22/2018
- CARD 18-27188; MSIV Line B Exceeded its Repair Guideline Penetration X-7B; 09/23/2018
- CARD 18-27189; RB1 Steam Tunnel/DW; 09/23/2018
- CARD 18-27229; MCC Position for e2150F005B has Loose Fuse Clips; 09/24/2018
- CARD 18-27774; Packing Leak from E2150F005B; 10/04/2018
- CARD 18-27941; Stem on Division 2 INBD ISO MOV; 10/07/2018
- CARD 18-28038; Packing Leak Observed from Division 2 Core Spray Inboard Isolation Valve E2150F005B During LLRT; 10/09/2018
- CARD 18-28188; C1100F010 Leakage Above Repair Guideline; 10/12/2018
- CARD 18-28190; C1100F181 Failed Local Leak Rate Test; 10/12/2018
- CARD 18-28253; Control Rod 26-43 Full in Light Goes Out When Lifting at Position 00; 10/13/2018
- CARD 18-28289; Upon Actuator Removal Discovered Water in the Compensator Assembly; 10/14/2018
- CARD 18-28311; Control Rod 14-27; 10/14/2018
- CARD 18-28349; CR 30-15 Became Uncoupled When Fully Withdrawn; 10/15/2018
- CARD 18-28349; CR 30-15 Became Uncoupled When Fully Withdrawn; 10/15/2018
- CARD 18-28350; Leakage Identified During RPV Pressure Test E1100F050B; 10/15/2018
- CARD 18-28351; RPV Flange Temperature Fluctuating from 130F to 140F on B21R007; 10/15/2018
- CARD 18-28352; Leakage Identified During RPV Pressure Test G3352F220; 10/15/2018
- CARD 18-28353; Leakage Identified During RPV Pressure Test B2100F026A; 10/15/2018
- CARD 18-28354; Multiple Rod Movement Message Received on RWM-OD During Performance of Scram Time Testing; 10/15/2018
- CARD 18-28355; Leakage Identified During RPV Pressure Test E1100F050A; 10/15/2018
- CARD 18-28356; Leakage Identified During RPV Pressure Test E1150F608; 10/15/2018
- CARD 18-28358; Leakage Identified During RPV Pressure Test E1100F116A; 10/15/2018
- CARD 18-28359; Leakage Identified During RPV Pressure Test E1150F009; 10/15/2018
- CARD 18-28360; Leakage Identified During RPV Pressure Test E1100F060A; 10/15/2018

- CARD 18-28361; Leakage Identified During RPV Pressure Test E1150F015B; 10/15/2018
- CARD 18-28362; Leakage Identified During RPV Pressure Test B3100F001A; 10/15/2018
- CARD 18-28364; Leakage Identified During RPV Pressure Test B3100F020; 10/15/2018
- CARD 18-28367; Leakage Identified During RPV Pressure Test CRDMs and LPRM; 10/16/2018
- CARD 18-28443; Control Rod Over-Travel for Rod 26-19; 10/17/2018
- CARD 18-28448; Control Rod 30-43 Missing 18 Indication; 10/17/2018
- CARD 18-29057; 2018 LIR: Analytical Low Margin Inboard MSIVs; 11/05/2018
- CARD 18-29096; E1150F068B Failed to Indicate Closed While Shutting Down Division 2 RHRSW; 11/06/2018
- EFA-E21-18-003; E2150F005B Packing Leak in Modes 4 and 5; Revision 0
- Procedure 23.208; RHR Complex Service Water Systems; Revisions 117 and 122
- Procedure 24.106.02; CRD Coupling Integrity Verification; Revision 34
- Procedure 24.205.06; Division 2 RHRSW Pump and Valve Operability Test; Revision 52
- Procedure 24.307.34; DGSW, DFOT and Starting Air Operability Test – EDG 11; Revision 55
- Procedure 24.307.35; DGSW, DFOT and Starting Air Operability Test – EDG 12; Revision 57
- Procedure 27.106.05; Control Rod Drive Timing Test and Adjustment; Revision 21
- Procedure 43.000.005; Visual Examination of Piping and Components (VT-2); Revision 36
- Procedure 43.401.311; Local Leakage Rate Testing for Penetration X-16A; Revision 26B
- Procedure 43.401.513; Core Spray Pressure Isolation Valve Leakage Test; Revision 34B
- Procedure 82.000.04; Refueling and Core Post – Alteration Verification; Revision 51
- RF19 Fermi Shiftly Outage Report; 10/12/2018
- TE-B21-18-024; Evaluation of the MSIVs N2 Pneumatic Supply Pressure; Revision 0
- WO 31886104; Final 43.401.311 LLRT for X-16A (Test-2:E2150F0085B); 09/24/2018
- WO 47472887; Final 43.401.500 LLRT for X-7D B2103F022D & F028D; 9/24/2018
- WO 47482772; Perform 24.106.02 CRD Coupling Integrity; 10/11/2018
- WO 49001468; Remove TM 17-0024 and Replace Valve Bonnet and Stem E1151F068B; 09/25/2018
- WO 50277733; Perform 24.307.34 Section EDG 11 DGSW Operability Test; 10/15/2018
- WO 50277751; Perform 24.307.35 Section 5.1 for PMT in RFO (EDG 12); 9/24/2018
- WO 52058671; Inspect/Replace Stem on Division 2 INBD ISO MOV; 10/14/2018
- WO 52307618; Valve E1150F068B Internal Inspection Repairs; 11/03/2018
- WO 52341199; Install TM 18-0030 (Remove E1150F068B Valve Internals); 11/08/2018

71111.20—Refueling and Other Outage Activities

- CARD 18-27788; RF-19 LL-Difficulty Encountered Viewing Skimmer Surge Tank Level Camera; 10/04/2018
- CARD 18-28219; Fatigue Management – Covered Work Performed by Uncovered Individual; 10/12/2018
- CARD 18-28400; NRC Question Braided Hose To/From Reactor Recirculation Pump B; 10/16/2018
- CARD 18-28574; NRC Identified Items in Drywell Needing Attention Before Close Out; 10/20/2018
- CARD 18-28654; As-Found Packing Less Than Adequate on E1100F050B; 10/23/2018
- CARD 18-28665; MSIVs Missing Heat Shields; 10/23/2018
- CARD 18-28666; Foreign Material Left in Drywell; 10/23/2018
- CARD 18-28743; Fatigue Management – Work Hours Not Tracked for Covered Worker in Accordance with MGA17 “Working Hour Limits” (Closed); 10/25/2018
- CARD 18-29675; Main Generator Rotor Ground Fault Alarm; 12/02/2018

- CARD 18-29827; Reactor Water Level Exceeded Level 8 Following Reactor Scram; 12/07/2018
- CARD 18-30361; Mispositioned Component Ops Shift 2 CLO, HU Error During Control Rod Exercising in Mode 4; 12/24/2018
- Estimated Critical Position Report; 10/24/2018
- Estimated Critical Report 12/30/2018
- MOP05-200; RPV Water Inventory Control; Revision 0
- MOP19; Reactivity Management; Revision 26A
- MWC13; Outage Nuclear Safety; Revision 18
- Procedure 20.000.21; Reactor Scram; Revision 68
- Procedure 22.000.01; Plant Start Up Master Checklist; Revision 72
- Procedure 22.000.02; Plant Startup to 25% Power; Revision 101
- Procedure 22.000.03; Power Operation 25% to 100% to 25%; Revision 103A
- Procedure 22.000.04; Plant Shutdown From 25% Power; Revision 85
- Procedure 22.000.05; Pressure/Temperature Monitoring During Heat Up and Cooldown; Revision 50
- Procedure 23.109; Turbine Operating Procedure; Revision 94
- Procedure 23.118; Main Generator and Generator Excitation; Revision 67
- Procedure 23.425.01; Primary Containment Procedures; Revision 79
- Procedure 23.623; Reactor Manual Control System; Revision 73
- Procedure 24.000.02; Shiftly, Daily, and Weekly Required Surveillances; Revision 153
- Procedure 24.000.05; Monthly Continuity Light and Channel Check; Revision 49A
- Procedure 54.000.07; Core Performance Parameter Check; Revision 59
- Procedure ARP 4D109; Generator Field Ground; Revision 10
- Reactivity Maneuvering Plan; BOC20 Reactor Startup; Revision 0
- Reactivity Maneuvering Plan; FO 18-02 Reactor Start Up; Revision 0
- Rod Pull Sheet; 10/05/2018
- Temporary Change Notice T-12683; Shutdown Margin Check; Revision 35
- Underwater Engineering Services, Inc. Quality Assurance Records Index; Torus Underwater Desludge, IWE/Coatings Inspection and Coating Repair
- WO 47427779; HPCI Suppression Pool Suction Strainer Damage; 04/04/2017
- WO 48693770; Extensive Coating Defects Identified During Torus Underwater Inspection; 09/18/2017
- WO 48693800; Inspect and Repair Torus Coating, Above Water; 09/18/2017

71111.22—Surveillance Testing

- CARD 18-26449; AIM to Track Procedure Changes Related to LOP/LOCA Frequency Extension; 08/27/2018
- CARD 18-27175; Typographical Errors Found During Performance of 44.030.002 Revision 40
- CARD 18-27179; White Trip Lights for G1103C037A/B Did Not Illuminate During 44.030.002 Revision 40 Step 6.3.29 and 6.3.30; 09/22/2018
- CARD 18-27183; Procedure 44.030.002 Needs Enhancement; 09/22/2018
- CARD 18-27187; Procedure 44.030.002 Needs Enhancement; 09/23/2018
- CARD 18-27258; NRC Concerns During Observation of Division 2 130/2620 Battery Testing; 09/24/2018
- CARD 18-27327; Incorrect Information Signed for in 42.309.06 Division 2 Battery Testing of 2B-1. Battery Cell Physical Damage Signed as Sat When Cells have Physical Damage; 09/25/2018
- Drawing 6M721-5707; Core Spray System; Revision AG

- Procedure 24.307.01; Emergency Diesel Generator 11 – Loss of Offsite Power and ECCS Start With Loss of Offsite Power Test; Revision 45
- Procedure 24.307.38; Simultaneous ECCS Start of Four Emergency Diesel Generators; Revision 32
- Procedure 42.309.06; Division 2 18 Month 130/260 VDC Battery Check; Revision 38
- Procedure 43.401.300; Local Leakage Rate Test Type C – General; Revision 55B
- Procedure 44.030.002; ECCS – Core Spray System, Division 2, Logic Functional Test; Revisions 40 and 40A
- RF19 Fermi Shiftly Outage Report; 10/19/2018
- Temporary Change Notice T-12684; Emergency Diesel Generator 11 – Loss of Offsite Power and ECCS; Revision 49
- WO 31886104; Final 43.401.311 LLRT for X-16A (Test – 2:E2150F005B); 9/24/2018

<u>71114.04—Emergency Action Level and Emergency Plan Changes</u>

- 10CFR50.54(q) Evaluation Number 2017-03E; May 9, 2017
- 10CFR50.54(q) Screen Number 2017-10S; May 9, 2017
- 10CFR50.54(q) Screen Number 2017-13S; March 17, 2017
- EP-101; Classification of Emergencies; Revisions 42 and 42B
- EP-590; 10CFR50.54(q) Screens and Evaluation; Revision 0
- Fermi 2 Radiological Emergency Response Preparedness Plan; Revision 47

<u>71124.01—Radiological Hazard Assessment and Exposure Controls</u>

- 67.000.101; Special Radiological Surveys and Analysis; Revision 48
- 68.000.002; Radiation and Contamination Surveys; Revision 0
- 68.000.005; Access Controls For High Radiation Areas; Revision 0A
- CARD 18-27507; Worker Received ED Dose Rate Alarm In RB1 Steam Tunnel
- CARD 18-27718; Tri-Nuke Filters Stored on Drywell Bulkhead
- CARD 18-2779; Current Radiological Data Not Used to Brief for Entry Into RWCU HX Room
- MRP06; Radiation Protection Conduct Manual; Revision 17A
- Radiological Surveys of RWCU HX Room; Various Dates 10/17/2013 – 10/1/2018
- RWP 182001; Operations/Fire Protection Tasks; Revision 00
- RWP 182025; G33 (RWCU), P73 (HWC) System Maintenance & Inspection; Revision 00
- RWP 183016; MSIVs – Drywell & RB Steam Tunnel – Maintenance & Inspections; Revision 00
- RWP 183019; B21/E21 System Maintenance & Inspection – Drywell & RB Steam Tunnel; Revision 00
- RWP 184001; RB5 – Reactor Disassembly & Support Activities; Revision 00
- RWP Radiological Pre-Job Briefing Form for RWCU HX Room Entry; 09/28/2018
- Self-Reading Dosimeter Alarm Evaluations; 09/28/2018 – 09/29/2018

<u>71124.02—Occupational As Low As Reasonably Achievable Planning and Controls</u>

- ALARA Package for RWP 182001; Operations/Fire Protection Tasks; Revision 00
- ALARA Package for RWP 182025; G33 (RWCU), P73 (HWC) System Maintenance & Inspection; Revision 00
- ALARA Package for RWP 183016; MSIVs – Drywell & RB Steam Tunnel – Maintenance & Inspections; Revision 00
- ALARA Package for RWP 183019; B21/E21 System Maintenance & Inspection – Drywell & RB Steam Tunnel; Revision 00

- ALARA Package for RWP 184001; RB5 – Reactor Disassembly & Support Activities; Revision 00

71151—Performance Indicator Verification

- RCS Specific Activity Performance Indicator Submittals; 10/01/2017-06/30/2018
- Occupational Exposure Control Effectiveness Submittals; 10/01/2017-06/30/2018
- RETS/ODCM Radiological Effluent Occurrences Submittals; 10/01/2017-06/30/2018

71152—Problem Identification and Resolution

- CARD 18-00146; Oil Detected Light On for Y4100P033; 09/13/2018
- CARD 18-23890; N30R827 Pt. 19 Steam Chest 4 Inner Surface Indicated Burnout in the MCR; 05/15/2018
- CARD 18-24053; Shift #2 to Perform CLO on Incorrect Operability Assessment for CARD 18-23890; 05/21/2018
- CARD 18-24133; 24/48 Cell 8 Low Voltage; 05/24/2018
- CARD 18-24184; 43.000.003 References Incorrect Division for Snubber T46-3093-G11; 05/25/2018
- CARD 18-24354; Oil Leak from Flange Supplying EDG11 LO Filter; 06/01/2018
- CARD 18-24453; Excessive Oil Leak from Flange on Discharge of Standby LO Pump; 06/01/2018
- CARD 18-24509; Cracked Lids on Division 2 Batteries; 06/07/2018
- CARD 18-25018; 43.000.003 References Incorrect Division for Snubbers T46-3092-G05, T46-3093-G10, G18A and B; 06/28/2018
- CARD 18-25136; Operability Assessment for G11R703
- CARD 18-25396; CTG 11-1 Start Following Maintenance; 07/14/2018
- CARD 18-25397; Trip of Number 1 Peaker Fuel Oil Forwarding Pump; 07/14/2018
- CARD 18-25732; NRC Identified Issue of a Declining Trend in Initial Shift Manager Operability Determination Cards; 07/27/2018
- CARD 18-26988; NRC Concern CARD 18-00146 Incorrect Documentation; 09/17/2018
- CARD 18-28245; C RHR Pump Failed to Start During Surveillance Testing; 10/13/2018
- CARD 18-28332; PM E784 Did Not Pass the PMT Step But Spec Sheet Was SAT; 10/15/2018
- CARD 18-28347; During Performance of 44.220 Step 6 1.29.2, B21-F506 Closed Indications Did Not Illuminate for Sub Steps A, B, and C. Acceptance Criteria Step 6 1.29.2.d Was Met With No Leakage; 10/15/2018
- CARD 18-28357; Excess Flow Check Valve B21F517C Open Light Failed to Actuate Properly; 10/15/2018
- CARD 18-28457; RHR Pump ‘C’ Tripped After Auto Start During 24.307.02; 10/18/2018
- CARD 18-28495; RHR Pump ‘A’ Failed to Start as Expected During 24.307.02; 10/18/2018
- CARD 18-28572; Degraded Grid Relaying is Not Bypassed when the EDGs are Supplying their Respective ESF Bus; 10/20/2018
- CARD 18-28795; Steam Leak Identified on E4150F003; 10/26/2018
- CARD 18-28990; Root Cause Evaluation Identified that EDP 80065 Inadvertently Changed 4160V ‘6’ Close Permissive; 11/01/2018
- CARD 18-30069; NRC Identified Potential Trend – Repetitive Failures of Important Equipment; 12/13/2018
- CARD 18-30293; Potential Knowledge Gap Associated with Seismic Monitoring Operability and Associated Reportability Requirements; 12/20/2018
- Drawing 6I721-2201-01; Residual Heat Removal Pump ‘A’ E1102C002A; Revision T
- Drawing 6I721-2572-15; 4160V ESS Bus “64C” – POS “C6”; Revision U

- Drawing 6I721-2572-28; 4160V ESS Buses 64B and 64C Load Shedding Strings; Revision U
- Drawing 6I721-2578-07; Relaying and Metering Diagram 4160V ESS. Bus 64C; Revision P
- Drawing 6SD721-2500-01; Plant 4160V and 480V System Service; Revision BS
- EDP-80065; Permanent Plant Modification to Ensure that the 4160V ESF Bus Degraded Voltage Load Shed Logic is Inhibited when Offsite Power is Not Connected; Revision 0
- MES27, Engineering Support Conduct Manual – Chapter 27 – Verification of System Operability; Revision 18
- MQA11; Corrective Action Program; Revision 46A
- MQA11-100; Operability Determination Process; Revision 2A
- ODE-11; CARD Documentation Expectations; Revision 27
- Procedure 20.300.SBO; Loss of Offsite and Onsite Power; Revision 27
- Procedure 24.307.01; Emergency Diesel Generator 11 – Loss of Offsite Power and ECCS Start with Loss of Offsite Power Test; Revisions 44, 49, and 50
- Procedure 24.307.04; Emergency Diesel Generator 14 – Loss of Offsite Power and ECCS Start with Loss of Offsite Power Test; Revision 50
- TE-R14-18-088; Past Operability Review of Degraded Grid Relaying and RHR Pump Motor Start; Revision 0
- WO 52174704; EDP-80065, Modification to Bus 64B Load Shedding Ckt; 10/21/2018
- WO 52174747; EDP-80065, Modification to Bus 64C Load Shedding Ckt; 10/21/2018
- WO 52174748; EDP-80065, Modification to Bus 64E Load Shedding Ckt; 10/21/2018
- WO 52174749; EDP-80065, Modification to Bus 64F Load Shedding Ckt; 10/21/2018

71153—Follow-Up of Events and Notices of Enforcement Discretion

- AQP-0001; Control of DTE Energy Owned Switchyard, Transformers, and Peaker Equipment at Fermi; Revision 5
- AQP-0002; 120kV and 345kV Switchyards; Revision 6
- CARD 10-23266; Evaluate INPO SEN 283, Dual Unit Scram with Equipment Complications for Impact to Fermi; 04/19/2010
- CARD 18-23026; Reactor Scram Due to Loss of 64 Transformer; 04/14/2018
- CARD 18-23052; Moisture Found Inside Bus 1-2B Breaker A6 Rear Cubicle; 04/15/2018
- CARD 18-26406; Roof Ridge Line on Bus 1-2A and Bus 3-4A Enclosures is Allowing Moisture to Enter the Enclosure; 08/24/2018
- CE Report CARD 18-23054; 8-Hour NRC Notification Missing Required Information; Revision 0
- Commercial Change Package 60041; Partial Sealing of Outdoor Metal-Clad Switchgear Roof Vent Cap; Revision 0
- Commercial Change Package 60042; Installation of Roof Flashing on Bus Ducts Penetrating Metal-Clad Switchgear, No. 1-2B Switchgear Building Roof; Revision 0
- Commercial Change Package 60044; Drilling Weep holes on Outdoor Metal-Clad Switchgear Roof; Revision 0
- MES83-107; Metal Enclosed Bus Inspection Program; Revision 1
- MLS04; Operating experience Program; Revisions 31 and 32
- MMR Appendix E; Maintenance Rule SSC Specific Functions; Revision 23A
- MMR14; Structures Monitoring; Revisions 4 and 5A
- Procedure 27.000.05; Operator Rounds; Revision 34
- Root Cause Evaluation Report; Reactor Scram Due to Loss of 64 Transformer; Revision 0