January 29, 2004

Mr. William O'Connor, Jr.
Vice President
Nuclear Generation
Detroit Edison Company
6400 North Dixie Highway
Newport, MI 48166

SUBJECT: ENRICO FERMI, UNIT 2
NRC INTEGRATED INSPECTION REPORT 05000341/2003010

Dear Mr. O'Connor:

On December 31, 2003, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Enrico Fermi, Unit 2. The enclosed report documents inspection findings which were discussed on January 9, 2004, with you, Mr. Cobb, and other members of your staff.

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

Based on the results of this inspection, two self-revealing findings of very low safety significance were identified, one of which involved a violation of NRC requirements. In addition, one issue which was reviewed under the NRC traditional enforcement process was determined to be a Severity Level IV violation of NRC requirements. However, because these violations were of very low safety significance and the issues were entered into your corrective action program, the NRC is treating these findings and issue as Non-Cited Violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest the subject or severity of a Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U. S. Nuclear Regulatory Commission - Region III, 801 Warrenville Road, Lisle, IL 60532-4351; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Fermi 2 facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

**/RA/**

Mark A. Ring, Chief
Projects Branch 1
Division of Reactor Projects

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

Enclosure: Inspection Report 05000341/20003010
w/Attachment: Supplemental Information

cc w/encl: N. Peterson, Director, Nuclear Licensing
P. Marquardt, Corporate Legal Department
Compliance Supervisor
R. Whale, Michigan Public Service Commission
L. Brandon, Michigan Department of Environmental Quality
Monroe County, Emergency Management Division
Planning Manager, Emergency Management Division
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REGION III

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| Docket No: | 50-341 |
| License No: | DPR-43 |
| Report No: | 05000341/2003010 |
| Licensee: | Detroit Edison Company |
| Facility: | Enrico Fermi, Unit 2 |
| Location: | 6400 N. Dixie Hwy.<br>Newport, MI 48166 |
| Dates: | October 1 through December 31, 2003 |
| Inspectors: | S. Campbell, Senior Resident Inspector<br>T. Steadham, Resident Inspector<br>R. Alexander, Radiation Specialist<br>E. Brown, Project Manager<br>R. Jickling, Emergency Preparedness Analyst<br>C. Phillips, Senior Operations Engineer (Lead Inspector)<br>D. Schrum, Reactor Engineer<br>P. Young, Examiner |
| Approved by: | M. Ring, Chief<br>Branch 1<br>Division of Reactor Projects |

Enclosure

## SUMMARY OF FINDINGS

IR 05000341/2003010; 10/01-12/31/2003; Fermi Nuclear Power Station, Unit 2; Surveillance Testing, Other Activities, and Emergency Preparedness.

This report covers a 3-month period of baseline resident inspection and announced baseline inspections on heat sink performance, licensed operator requalification, emergency preparedness, and radiation safety. The inspection was conducted by Region III inspectors and the resident inspectors. One Severity Level IV Non-Cited Violation (NCV) and two Green findings involving one NCV were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be "Green" or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

### A. <u>Self-Revealing Findings</u>

#### Cornerstone: Mitigating Systems

Green. A finding of very low safety significance was self-revealed on November 25, 2003, when a technician failed to follow procedures and improperly connected test equipment to the wrong terminals to perform a reactor core isolation cooling (RCIC) steam diaphragm pressure test. This caused a short circuit and an unexpected steam isolation of the RCIC system.

This finding was more than minor because RCIC was rendered inoperable and unavailable for about 3 hours, thereby affecting the availability and reliability of a mitigating system as described in the mitigating systems cornerstone. The finding was of very low safety significance because RCIC could be quickly restored by opening RCIC steam supply outboard containment isolation valve E5150F008. Thus, the safety function of providing high pressure water to the core in the event of a loss of feedwater was not lost. This failure to follow procedures was a violation of 10 CFR 50, Appendix B Criterion V and is classified as a Non-Cited Violation. (Section 1R22.2)

Green. A finding of very low safety significance was self-revealed during the August 14, 2003, blackout when station blackout combustion turbine generator (CTG) 11-1 failed to start. An improper modification process used in 1996 to install an inverter on CTG 11-1 did not include updating the design basis central component (CECO) database with the appropriate low voltage inverter trip set point. The low voltage trip set point was set too high and prevented CTG 11-1 from starting on demand during the blackout.

This finding was more than minor because it affected the mitigating systems cornerstone objective to ensure availability, reliability, and capability of the CTG 11-1 system that responds to initiating events to prevent undesirable consequences. The issue was of very low safety significance because the inspectors answered “no” to all five screening questions in the Phase 1 Screening Worksheet under the mitigating systems column. Since CTG 11-1 was a non-Technical Specification system and not required by 10 CFR Part 50, Appendix B, no violation of regulatory requirements occurred. (Section 4OA5.1)

**Cornerstone: Emergency Preparedness**

Severity Level IV. The inspectors identified that the licensee changed its standard emergency action level (EAL) scheme on December 19, 2000, for those events related to toxic gas releases for Unusual Event and Alert classifications. The inspectors determined these changes decreased the effectiveness of the emergency plan, and the licensee did not obtain prior NRC approval, contrary to the requirements of 10 CFR 50.54(q).

Because the issue affected the NRC's ability to perform its regulatory function, it was evaluated with the traditional enforcement process as specified in Section IV.A.3 of the Enforcement Policy. According to Supplement VIII of the Enforcement Policy, this issue was determined to be a Severity Level IV violation because it involved a failure to meet a requirement not directly related to assessment and notification. Further, this problem was isolated to two EALs and was not indicative of a functional problem with the EAL scheme. Because the licensee has entered this issue into its corrective action program it is being treated as a Non-Cited Violation. (Section 1EP4)

**B. <u>Licensee-Identified Violations</u>**

No findings of significance were identified.

## REPORT DETAILS

**Summary of Plant Status**

Fermi 2 began this inspection period at approximately 90 percent rated thermal power due to both control center heating, ventilation and air conditioning (CCHVAC) chillers being declared inoperable with subsequent entry into Technical Specification 3.0.3. At 12:15 a.m. on October 1, 2003, the Division 1 CCHVAC chiller was declared operable which allowed the licensee to exit the unit shutdown. The reactor reached full rated thermal power at 12:58 a.m. on October 1, 2003, where it remained until December 6, 2003. At 1:10 a.m. on December 6, 2003, reactor power was decreased to approximately 65 percent to support turbine bypass valve testing, scram time testing, and rod pattern adjustments. At 9:25 a.m. later that day, reactor power was returned to 100 percent where it remained for the remainder of the inspection period.

### 1. REACTOR SAFETY

**Cornerstone: Mitigating Systems**

1R01 Adverse Weather (71111.01)

.1 Cold Weather Preparations

a. Inspection Scope

The inspectors selected three risk-significant systems (residual heat removal (RHR) complex, combustion turbine generator (CTG) 11-1, and the general service water (GSW) complex) that are required to be protected from adverse cold weather. The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR), Technical Specifications, adverse weather procedures, and other plant documents to determine that the systems or components will remain functional when challenged by cold weather conditions.

b. Findings

No findings of significance were identified.

.2 High Winds

a. Inspection Scope

The inspectors toured the plant and used the Acts of Nature Abnormal Operating Procedure and the UFSAR to verify that the licensee was prepared for seasonal readiness related to:

- The RHR complex for the mechanical draft cooling tower fans;
- The spent fuel pool blowoff panels;

- The reactor building superstructure steel frame; and,
- The reactor building crane.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04Q)

.1 Partial Walkdowns

a. Inspection Scope

The inspectors performed five partial walkdowns of accessible portions of risk-significant, mitigating systems during times when the systems were of increased importance due to redundant divisions or other related equipment being unavailable. The walkdowns were performed to verify proper alignment of valves, control switches and clear control room annunciator alarms. The inspectors reviewed associated piping and instrumentation drawings, condition assessment resolution documents (CARDs) and used the system operating procedures lineup to verify the system standby alignment. The inspectors used the documents to determine standby readiness of the system. The inspectors reviewed the following systems:

- Reactor core isolation cooling (RCIC);
- Division 1 RHR service water (RHRSW) system;
- CTG 11-1;
- Division 1 engineered safety feature dc power system;
- Emergency diesel generator (EDG) 11.

b. Findings

No findings of significance were identified.

1R05 Annual Fire Drill Observation (71111.05A)

a. Inspection Scope

On November 21, 2003, the inspectors observed the fire brigade respond to a simulated oil fire on the first floor of the turbine building to evaluate the readiness of licensee personnel to fight fires.

b. Findings

No findings of significance were identified.

1R06 Flood Protection (71111.06)

a. Inspection Scope

The inspectors reviewed the licensee's procedures for internal flooding as well as the UFSAR internal flooding analysis. The inspectors walked down areas associated with internal flooding in both the reactor and turbine buildings. The inspectors also performed a walkdown of both Division 1 and 2 RHR pump corner rooms to check floor drains and sumps for conditions that could create or exacerbate an internal flood hazard condition.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

.1 Biennial Review of Heat Sink Performance

a. Inspection Scope

The inspectors verified that the licensee's maintenance, testing, inspection and evaluation of results were adequate to ensure proper heat transfer for the following heat exchangers:

- The RHR heat exchangers with residual heat removal service water (RHRSW) as the cooling water source;
- The emergency equipment cooling water (EECW) heat exchangers with emergency equipment service water (EESW) as the cooling water source.

The inspectors reviewed heat exchanger test methodology, frequency of testing, test conditions, acceptance criteria and trending of results. The inspection, cleaning, and maintenance methods used to evaluate the RHR/RHRSW and EECW/EESW systems' reliability were reviewed with technical specialists. This review was to verify that the methods used for inspection and cleaning were consistent with expected degradation and that the final condition of heat exchangers was acceptable. This review included reviewing operability evaluations for some degraded conditions. In addition, the inspectors reviewed and compared plant conditions to the commitments that the licensee had made in response to Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

Selected test calculations of component performance data were reviewed to verify the test results reflected heat exchanger conditions and that operation was consistent with design. The inspectors reviewed the design basis documents for the RHR and RHRSW and compared plant conditions to design information. In addition, the inspectors assessed the trending of the measured data for the components inspected and the licensee's proposed actions for results not within the acceptance criteria.

The service water chemical treatment program for the ultimate heat sink (UHS) was reviewed and discussed with chemistry and engineering personnel to verify that: (1) potential biofouling mechanisms had been identified; (2) treatments were conducted as scheduled; and (3) results were monitored for effectiveness.

The inspectors reviewed a sample of condition assessment resolution documents (CARDs) related to selected equipment and programs to verify that the licensee had an appropriate threshold for identifying issues, entered them in the corrective action program, and that they were appropriately resolved.

b. Findings

No findings of significance were identified.

.2 Semiannual Review of Heat Sink Performance

a. Inspection Scope

The inspectors reviewed the RHR Division 2 mechanical seal heat exchanger. The inspectors determined if deficiencies masked degraded performance. Also, the inspectors reviewed corrective actions and engineering analysis regarding the low EECW flows to the coolers to determine if operability was justified. Finally, the inspectors verified that the licensee had adequately identified and resolved heat sink performance problems (corrosion, fouling, silting) that could result in affecting heat exchanger performance. A specific evaluation of the prioritization and evaluation of corrective actions associated with these seal coolers is discussed in Section 4OA2.2, "Corrective Actions Involving Calculations and Engineering Functional Analyses," of this report.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

.1 Facility Operating History

a. Inspection Scope

The inspectors reviewed the plant's operating history from November 2001, through November 2003, to assess whether the Licensed Operator Requalification Training (LORT) program had addressed operator performance deficiencies noted at the plant.

b. Findings

No findings of significance were identified.

.2 Licensee Requalification Examinations

a. Inspection Scope

The inspectors performed a biennial inspection of the licensee's LORT program. The inspectors reviewed the annual requalification operating and biennial written examination material to evaluate general quality, construction, and difficulty level. The examination material reviewed consisted of two dynamic simulator scenarios, five job performance measures (JPMs), and two biennial written examinations (one Reactor Operator (RO) and one Senior Reactor Operator (SRO)) consisting of 32 open reference multiple choice questions. The inspectors reviewed the methodology for developing the examinations, including the LORT program 2-year sample plan, probabilistic risk assessment insights, previously identified operator performance deficiencies, and plant modifications. The inspectors also reviewed and assessed the level of examination material duplication within the current year annual examinations and to the previous year's examinations. Additionally, the inspectors interviewed members of the licensee's management, operations, and training staff and discussed various aspects of the examination development.

b. Findings

No findings of significance were identified.

.3 Licensee Administration of Requalification Examinations

a. Inspection Scope

The inspectors observed the administration of the requalification operating test to assess the licensee's effectiveness in conducting the test and to assess the facility evaluators' ability to determine adequate performance using objective and measurable performance standards. The inspectors evaluated the performance of one shift crew in parallel with the facility evaluators during two dynamic simulator scenarios. In addition, the inspectors observed licensee evaluators administer thirteen JPMs to five licensed operators. The inspectors also observed the administration of the written examination to one operating crew. The inspectors evaluated the ability of the simulator to support the examinations. A specific evaluation of simulator performance was conducted and documented under Section 1R11.7, "Conformance With Simulator Requirements Specified in 10 CFR 55.46," of this report. The inspectors also reviewed the licensee's overall examination security program.

b. Findings

No findings of significance were identified.

.4 Licensee Training Feedback System

a. Inspection Scope

The inspectors assessed the methods and effectiveness of the licensee's processes for revising and maintaining its LORT program up to date, including the use of feedback from plant events and industry experience information. The inspectors interviewed licensee personnel (operators, instructors, training management, and operations management) and reviewed applicable licensee procedures. In addition, the inspectors reviewed the licensee's training department self-assessment reports, to evaluate the licensee's ability to assess the effectiveness of its LORT program and to implement appropriate corrective actions.

b. Findings

No findings of significance were identified.

.5 Licensee Remedial Training Program

a. Inspection Scope

The inspectors assessed the adequacy and effectiveness of the remedial training conducted since the previous annual requalification examinations and the training planned for the current examination cycle to ensure that they addressed weaknesses in licensed operator or crew performance identified during training and plant operations. The inspectors reviewed remedial training procedures and individual remedial training plans, and interviewed licensee personnel (operators, instructors, and training management). In addition, the inspectors reviewed the licensee's previous NRC annual examination cycle remediation packages for unsatisfactory operator performance on the operating test to ensure that remediation and subsequent re-evaluations were completed prior to returning individuals to licensed duties.

b. Findings

No findings of significance were identified.

.6 Conformance With Operator License Conditions

a. Inspection Scope

The inspectors reviewed the facility and individual operator licensees' conformance with the requirements of 10 CFR Part 55. The inspectors reviewed the facility licensee's program for maintaining active operator licenses and to assess compliance with 10 CFR 55.53 (e) and (f). The inspectors reviewed the procedural guidance and the process for tracking on-shift hours for licensed operators and which control room positions were granted credit for maintaining active operator licenses. The inspectors also reviewed six licensed operator medical records maintained by the facility's nurse and assessed compliance with the medical standards delineated in ANSI/ANS-3.4, "American National Standard Medical Certification and Monitoring of Personnel Requiring Operator

Licenses for Nuclear Power Plants," and with 10 CFR 55.21 and 10 CFR 55.25. In addition, the inspectors reviewed the facility licensee's LORT program to assess compliance with the requalification program requirements as described by 10 CFR 55.59 (c).

b. Findings

No findings of significance were identified.

.7 Conformance with Simulator Requirements Specified in 10 CFR 55.46

a. Inspection Scope

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

b. Findings

No findings of significance were identified.

.8 Biennial Written Examination and Annual Operating Test Results

a. Inspection Scope

The inspectors reviewed the overall pass/fail results of individual written tests, JPM operating tests, and simulator operating tests (required to be given per 10 CFR 55.59(a)(2)) administered by the licensee from October 13 through December 6, 2003. The overall results were compared with the significance determination process in accordance with NRC Manual Chapter 0609I, "Operator Requalification Human Performance Significance Determination Process (SDP)."

b. Findings

No findings of significance were identified.

.9 Quarterly Review of Licensed Operator Requalification Testing and Training Activities

a. Inspection Scope

On October 21, 2003, the inspectors observed an Operations support crew during the annual requalification examination in mitigating the consequences of events in Scenario No. SS-OP-904-0020, "SCRAM discharge volume vent & drain failure, loss of air, safety relief valve open, and loss of coolant accident" on the simulator. The inspectors evaluated the following areas:

- Licensed operator performance;
- Crew's clarity and formality of communications;
- Ability to take timely actions in the conservative direction;
- Prioritization, interpretation, and verification of annunciator alarms;
- Correct use and implementation of abnormal and emergency procedures;
- Control board manipulations;
- Oversight and direction from supervisors; and
- Ability to identify and implement appropriate Technical Specification actions and Emergency Plan actions and notifications.

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

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12Q)

a. Inspection Scope

The inspectors reviewed applicable system health reports, associated CARDs, licensee maintenance rule conduct manual, various surveillance tests, applicable design basis documents, maintenance rule scoping determinations, expert panel meeting notes, monthly monitoring reports, and the control room unit logs for the following systems:

- Standby Liquid Control (C4100);
- Standby Feedwater (N2103);
- Reactor Protection System (C7100).

The inspectors independently evaluated the licensee's determination of maintenance rule functional failures, reviewed surveillance procedures and operators' logs to assess licensee calculation of system unavailability. The inspectors also reviewed licensee-established performance goals and 'Get Well' programs for systems that do not meet performance goals or (a)(1) status systems.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed the documents listed in the "List of Documents Reviewed" section of this report to determine if the risk associated with the downpower on December 6, 2003, with the results provided by the licensee's risk assessment tool. The inspectors conducted walkdowns to ensure that redundant mitigating systems and/or barrier integrity equipment credited by the licensee's risk assessment remained available. When compensatory actions were required, the inspectors conducted plant inspections to validate that the compensatory actions were appropriately implemented. The inspectors also discussed emergent work activities with the shift manager and work week manager to ensure that these additional activities did not change the risk assessment results.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors assessed the following engineering functional analyses (EFA) to ensure that operability was properly justified and the component or system remained available such that no unrecognized increase in risk occurred:

- Air in Chiller Oil Sensing Line (CARD 03-21549 & EFA T41-03-016);
- Loss of NIAS to E41F035 (EFA-E41-02-007);
- SLO Multiplier not included in LHGR limits (EFA-J11-03-011, Rev A); and
- HPCI High Point Vent Valve 1/4 Open (EFA-E41-03-001).

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors performed a review of all operator workarounds and challenges identified as of December 1, 2003, to perform a semiannual risk assessment of revised operator workarounds. The inspectors also compared workaround information to the normal, abnormal, and emergency operating procedures to ensure that operations personnel maintained the ability to correctly respond to plant transients in a timely manner. The inspectors utilized system knowledge, reviewed plant procedures, and interviewed operations personnel to ensure that the workarounds and challenges previously identified did not adversely impact system reliability and availability, create the potential for system misoperation, or result in a workaround that impacted multiple mitigating equipment.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed and observed the following post-maintenance testing (PMT) activities involving risk significant equipment in the mitigating systems cornerstone:

- PMT on ISI valves;
- Standby feedwater test valve;
- Standby feedwater test valve internal replacement;
- PMT on replaced accumulators for control rods 06-16 and 42-11 and scram time testing;
- PMT on high pressure stop valve number 2 limit switch;
- PMT of reactor core isolation cooling after component repairs;
- PMT on repair of automatic depressurization system timer logic;
- PMT on repair of reactor water cleanup isolation valve (G3352F220);
- PMT on repair of reactor water cleanup isolation valve (G3352F004); and
- PMT on residual heat removal K35A relay.

The inspectors verified that the post-maintenance test was adequate for the scope of the maintenance work performed, acceptance criteria were clear, and operational readiness consistent with design and licensing basis documents was demonstrated. The inspectors also verified that the impact of the testing had been properly characterized in the risk assessment, the test was performed as written, the testing prerequisites were satisfied, and that the test data was complete. Following the completion of the test, the inspectors verified that the system was returned to its normal standby configuration.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 Routine Surveillance Testing

a. Inspection Scope

The inspectors observed surveillance testing or reviewed test data for risk-significant systems or components to assess compliance with Technical Specifications, 10 CFR 50 Appendix B, and licensee procedure requirements and evaluated the consistency of the tests with the UFSAR. The inspectors verified that the testing demonstrated that the systems were ready to perform their intended safety functions. Additionally, the inspectors reviewed whether test control was properly coordinated with the control room and performed in the sequence specified in the surveillance instruction and if test equipment was properly calibrated and installed to support the surveillance tests. The specific surveillance activities assessed were:

- High pressure coolant injection test performed during week of October 12, 2003;
- Standby feedwater test valve testing after internals replacement performed during the week of October 19, 2003;
- EDG 14 slow start surveillance performed on November 11, 2003;
- Standby gas treatment 10-hour run performed during the week of November 16, 2003;
- Reactor core isolation cooling quarterly test performed during the week of November 30, 2003; and
- Reactor recirculation limiter verification performed during the week of November 30, 2003;

b. Findings

No findings of significance were identified.

.2 RCIC Turbine Exhaust Diaphragm High Pressure Division 1 Functional Test

a. Inspection Scope

The inspectors reviewed schematic diagram I-2235-03, "RCIC System Logic Circuit, Part 2," job identification 0045031124 and interviewed instrumentation and control personnel to understand the circumstances that caused an inadvertent isolation of the RCIC system during an RCIC turbine high exhaust diaphragm pressure functional test.

b. Findings

Introduction: One Green NCV was self-revealed for failure to comply with 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," having very low safety significance for not following Procedure 44.020.243, "NSSSS-RCIC Turbine Exhaust Diaphragm Pressure, Division 1 Functional Test." Improperly installing measuring and test equipment (M&TE) across the incorrect logic terminals caused RCIC to isolate unexpectedly, rendering it unavailable, on November 25, 2003.

Description: On November 25, 2003, an instrumentation and control technician performed Procedure 44.020.243, "NSSSS-RCIC Turbine Exhaust Diaphragm Pressure, Division 1 Functional Test," without an independent verifier. Step 6.1.5 of the procedure required the technician to actuate the trip setpoint for turbine exhaust diaphragm high pressure trip unit E51-N655A at the H21P080 testability panel. This action closed contact E51K201A in the trip unit. Step 6.1.7 directed a short circuit reading be made across this contact by connecting M&TE at terminals AA-50 and AA-52. Instead, the technician incorrectly connected the M&TE across contact E51K201C (terminal AA-51 vice terminal AA-52) that was in series with contact E51K201A.

As the technician dialed through the ac, dc, and ohm readings on the M&TE, he missed that a dc reading was present thereby indicating that E51K201A was closed. When he moved the dial to the ohm reading, a short circuit around contact E51K201C completed the logic to isolate the RCIC steam supply outboard containment isolation valve E5150F008 and actuate alarm 1D51, "RCIC Isolation Trip Signal A." Closing this valve caused RCIC to become unavailable unexpectedly . Operations personnel stopped the test to conduct an investigation and initiated CARD 03-23171. As a result of this event, RCIC was unavailable for approximately 3 hours.

Analysis: The inspectors determined that connecting the M&TE to the wrong terminals was a performance deficiency warranting a significance evaluation in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," issued on June 20, 2003. The inspectors determined that the finding was more than minor because the unplanned unavailability of the RCIC system affected an attribute described under mitigating systems cornerstone of the reactor oversight process for availability.

Using Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors answered "no" to all five screening questions in the Phase 1 Screening Worksheet under the mitigating systems column. Specifically, since the isolation of the RCIC could be quickly restored by opening RCIC steam supply outboard containment isolation valve E5150F008, the safety function of providing high pressure water to the core in the event of a loss of feedwater was not lost. The inspectors therefore concluded that the issue was of very low safety significance.

Enforcement: 10 CFR Part 50, Appendix B, Criterion V, states, in part, that activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these procedures. Contrary to the above, an instrumentation and control technician failed to follow Procedure 44.020.243, "NSSSS-RCIC Turbine Exhaust Diaphragm Pressure, Division 1 Functional Test," Step 6.1.7 and installed a jumper around contact E51K201C instead of E51K201A while contact E51K201A was closed. Consequently, the installation of the jumper completed the logic to isolate RCIC steam supply outboard containment isolation valve E5150F008, rendering RCIC unavailable for approximately 3 hours. Because the failure to follow the test procedure is of very low safety significance and has been entered into the licensee's corrective action program (CARD 03-23171), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy **(NCV 05000341/2003010-01)**.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed documentation for temporary configuration change for Temporary Modification 03-0022, "temporary pump in D076 sump."

The inspectors assessed the acceptability of the temporary configuration change by comparing 10 CFR 50.59 screening and evaluation information against the UFSAR and Technical Specifications. The comparison was performed to ensure that the new configuration remained consistent with design basis information. The inspectors verified that the modification was installed as directed; operated as expected; modification testing adequately demonstrated continued system operability, availability, and reliability, and that operation of the modification did not impact the operability of any interfacing system. The inspectors also reviewed condition reports initiated during or following temporary modification installation to ensure that problems encountered during installation were appropriately resolved.

b. Findings

No findings of significance were identified.

**Cornerstone: Emergency Preparedness**

1EP2 Alert and Notification System (ANS) Testing (71114.02)

a. Inspection Scope

The inspectors discussed with Emergency Preparedness (EP) staff the design, equipment, and periodic testing of the public ANS for the Fermi 2 reactor facility emergency planning zone to verify that the system was properly tested and maintained. The inspectors also reviewed procedures and records for a 15 month period ending September 2003, related to ANS testing, annual preventive maintenance, and non-scheduled maintenance. The inspectors reviewed the licensee's documentation for determining whether each model of siren installed in the emergency planning zone would perform as expected if fully activated. Records used to document and trend component failures for each model of installed siren were also reviewed to ensure that corrective actions were taken for test failures or system anomalies.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization (ERO) Augmentation Testing (71114.03)

a. Inspection Scope

The inspectors reviewed the licensees ERO augmentation testing to verify that the licensee maintained and tested its ability to staff the ERO during an emergency in a timely manner. Specifically, the inspectors reviewed bimonthly, off-hours staff augmentation test procedures, related May 30, 2001, through September 29, 2003, test records, primary and backup provisions for off-hours notification of the Fermi 2 reactor facility emergency responders, and the current ERO rosters for Fermi 2. The inspectors reviewed and discussed the facility emergency preparedness (EP) staff's provisions for maintaining ERO call-out lists.

b. Findings

No findings of significance were identified.

1EP4 <u>Emergency Action Level (EAL) and Emergency Plan Changes</u> (71114.04)

a. <u>Inspection Scope</u>

The inspectors conducted a review of EALs HU3 and HA3 located in the Radiological Emergency Response Preparedness Plan and implementing procedure EP-101, "Classification of Emergencies." These EALs pertain to releases of toxic and flammable gases at Fermi 2, and the review was performed to determine if the changes made decreased the effectiveness of the Radiological Emergency Response Preparedness Plan (E-Plan). The applicable portions of 10 CFR 50.54(q), 10 CFR 50.47(b), 10 CFR 50, Appendix E, and NUMARC/NESP-007, "Methodology for Development of Emergency Action Levels," were used as reference criteria.

b. <u>Findings</u>

<u>Introduction</u>

The licensee changed its EALs that address events related to toxic gases as a clarification for emergency response organization decision makers. These changes were determined to decrease the effectiveness of the emergency plan at Fermi 2. The licensee did not submit these changes to the NRC for prior approval. This is a violation of 10 CFR 50.54(q) and, because it impacts the regulatory process, traditional enforcement was applied. Because this item was entered into the licensee's corrective action program and because it involved a failure to meet a requirement not directly related to assessment and notification, this issue was determined to be a Severity Level IV Non-Cited Violation.

<u>Description</u>

On December 19, 2000, the licensee approved a revision of the implementing procedure EALs in EP-101, Revision 26, "Classification of Emergencies," pertaining to toxic gas events. These changes were, in part, intended as a clarification and were based upon the licensee's assessment that the certain conditions did not meet the intent of the NUMARC EALs. The revision to the licensee's standardized emergency action level scheme in their implementing procedures resulted from their evaluation of industry events in which Unusual Events and Alerts had been declared for carbon dioxide and Halon activations that they believed did not meet the intent for the toxic gas EALs.

Also, during training evolutions, operations staff questioned the toxic gas EALs and believed it was inappropriate to declare Unusual Events and Alerts for some conditions. Below are the previous EALs, the NUMARC EALs, and the revised EALs addressing toxic gas for the Unusual Event and Alert, respectively. The notes added to the Unusual Event and Alert EAL revisions were the clarifying changes made to these EALs.

<u>Previous Unusual Event EAL</u>
Report or detection of toxic or flammable gases that could enter within the site area boundary in amounts that can affect normal plant operations.

NUMARC Unusual Event EAL
Report or detection of toxic or flammable gases that could enter within the site area boundary in amounts that can affect normal operation of the plant.

Revision Unusual Event EAL
Report or detection of toxic or flammable gases that could enter within the site area boundary in amounts that can affect normal plant operations.

**NOTE (2):** Fire suppression gases are not considered toxic for the purpose of this EAL.

Previous Alert EAL
Report or detection of toxic gases within a facility structure in concentrations that will be life threatening to plant personnel.

NUMARC Alert EAL
Report or detection of toxic gases within or contiguous to a VITAL AREA in concentrations that may result in an atmosphere immediately Dangerous to Life and Health (IDLH).

Revised Alert EAL
Report or detection of toxic gases within a facility structure in concentrations that will be life threatening to plant personnel.

**NOTE (1):** Fire suppression gases are not considered toxic for the purpose of this EAL.

The NRC's review of this matter resulted in several observations. First, the change to the EALs by the addition of Note 2 for the Unusual Event and Note 1 for the Alert would reduce the number of declarable events by removing fire suppression gases (including Halon and carbon dioxide) from the gases that could affect normal plant operations. Second, the licensee's E-Plan implementing procedures were not consistent with the E-Plan in that the E-Plan did not include the notes stating that fire suppression gases were not considered toxic for purposes of these EALs. (With the revised implementing EALs, no emergency classification would be made for carbon dioxide or Halon activations in any areas or rooms in the plant). Based upon these considerations, the NRC concluded that the licensee's changes decreased the effectiveness of the emergency plans. Such changes are to be submitted to the NRC for review before implementation. The licensee did not submit changes to the NRC and was therefore in violation of 10 CFR 50.54(q).

Analysis

The inspectors determined that the licensee failed to meet the requirements of 10 CFR 50.54(q) when they failed to identify a decrease in effectiveness of its standard EAL classification scheme following the revision. A standard classification and action level scheme is required by 10 CFR 50.47(b)(4). No actual safety consequence was identified; however, the inspectors determined that the issue had a potential for impacting the NRC's ability to perform its regulatory function. As such, traditional enforcement was applied instead of the Significance Determination Process (SDP).

Enforcement

10 CFR 50.54(q) states, in part, that the "licensee may make changes to these plans without Commission approval only if the changes do not decrease the effectiveness of the plans. Proposed changes that decrease the effectiveness of the approved emergency plans may not be implemented without application to and approval by the Commission." On December 19, 2000, the licensee made changes to their standard EAL scheme in the E-Plan implementing procedures which reduced the effectiveness of the emergency plans. These changes were not submitted to the NRC for approval prior to implementation. The licensee entered this issue into their corrective action program (CARD 03-22478) and potential corrective actions include evaluating these EALs, other licensee's EALs, and contacting the industry group, NEI, to correct the wording of these EALs.

Changing emergency plan commitments without prior approval impacts the NRC's ability to perform its regulatory function and is therefore processed through traditional enforcement as specified in Section IV.A.3 of the Enforcement Policy, issued May 1, 2000 (65 FR 25388). According to Supplement VIII of the Enforcement Policy, this finding was determined to be a Severity Level IV violation because it involved a failure to meet a requirement not directly related to assessment and notification. Further, this problem was isolated to two EALs and was not indicative of a functional problem with the EAL scheme. Additionally, because the licensee has entered this issue into their corrective action program this finding is being treated as Non-Cited Violation (Severity Level IV) consistent with Section VI.A of the Enforcement Policy **(NCV 05000341/2003010-02)**.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies (71114.05)

a. Inspection Scope

The inspectors reviewed the Nuclear Assurance staff's 2002 and 2003 Audit Reports to ensure that this audit complied with the requirements of 10 CFR 50.54(t) and that the licensee adequately identified and corrected deficiencies. The inspectors also reviewed the EP staff's 2003 self-assessments, and critiques to evaluate the EP staff's efforts to identify and correct weaknesses and deficiencies. Additionally, the inspectors reviewed a sample of EP items, condition reports, and corrective actions related to the facility's EP program to determine whether corrective actions were acceptably completed.

Finally, the inspectors reviewed an actual emergency plan activation to determine if the licensee effectively implemented the requirements of the E-Plan.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed the licensee perform a simulator-based training evolution on November 25, 2003, that contributed to the “drill/exercise performance” performance indicator. The inspectors observed activities in the control room simulator and attended the post evaluation critiques in the simulator immediately following the evaluation. The focus of the inspectors’ activities was to note any weaknesses and deficiencies in operator performance, ensure that the licensee evaluators noted the same weaknesses and deficiencies, and entered any weaknesses or deficiencies into the corrective action program as appropriate. The inspectors placed emphasis on observations regarding event classification, notifications, and protective action recommendations. The inspectors reviewed scenario SS-OP-904-0017, “Fire impairment, loss of reactor building closed cooling water, loss of vacuum, and anticipated transient without SCRAM.”

b. Findings

No findings of significance were identified.

## 2. RADIATION SAFETY

### Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Review of Licensee Performance Indicators for the Occupational Exposure Cornerstone

a. Inspection Scope

The inspectors reviewed the licensee’s occupational exposure control cornerstone performance indicator (PI) to determine whether or not the conditions surrounding PI occurrences, if any, had been evaluated, and identified problems had been entered into the corrective action program for resolution. This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Plant Walkdowns and Radiation Work Permit (RWP) Reviews

a. Inspection Scope

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

- Turbine Building;
- Reactor Building (including all four corner rooms); and
- Onsite Storage Facility.

The inspectors reviewed routine and outage-related RWPs for work areas with the potential to become airborne radioactivity areas to verify barrier integrity and engineering control contingencies were in place and to determine if there was a potential for individual worker internal exposures of >50 millirem committed effective dose equivalent. Though no such areas were identified, the inspectors discussed with radiation protection (RP) staff those work areas having a history of, or the potential for, airborne transuranics to verify that the licensee had considered the potential for transuranic isotopes and provided appropriate worker protection.

The inspectors also reviewed the licensee's physical and programmatic controls for highly activated and/or contaminated materials (non-fuel) stored within spent fuel or other storage pools. These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed the licensee's self-assessments, audits, Licensee Event Reports, and Special Reports, as available, related to the access control program to verify that identified problems were entered into the corrective action program for resolution.

The inspectors evaluated the licensee's process for problem identification, characterization, prioritization, and verified that problems were entered into the corrective action program and resolved. For repetitive deficiencies and/or significant individual deficiencies in problem identification and resolution, the inspectors reviewed licensee self-assessment activities to verify that they were capable of identifying and addressing these deficiencies.

As available, the inspectors reviewed licensee documentation packages for all PI events occurring since the last inspection to determine if those PI events involved dose rates greater than 25 R/hour at 30 centimeters or greater than 500 R/hour at 1 meter. Barriers were evaluated for failure and to determine if there were any barriers left to prevent personnel access. As available, unintended exposure incidents of greater than 100 millirem total effective dose equivalent (or greater than 5 rem shallow dose equivalent or greater than 1.5 rem lens dose equivalent), were evaluated to determine if there were any regulatory overexposures or if there was a substantial potential for an overexposure. These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.4 Job-In-Progress Reviews

a. Inspection Scope

The inspectors observed the following work activity that was being performed in radiation areas, airborne radioactivity areas, or high radiation areas for observation of work activities that presented the greatest radiological risk to workers:

- Preparation and loading of a High Integrity Container of dewatered resin waste into a shipping cask as part of a Type B shipment to the Barnwell Disposal Facility in South Carolina.

The inspectors reviewed radiological job requirements for this activity including RWP requirements and work procedure requirements, and attended RP pre-job briefings. Job performance was observed with respect to these requirements to verify that radiological conditions in the work area were adequately communicated to workers through pre-job briefings and postings.

These reviews were not counted as inspections samples, but were additional reviews to augment the inspector's previous inspection of the licensee's radioactive waste and transportation programs previously documented in Section 2PS2 of Inspection Report 05000341/2003008.

b. Findings

No findings of significance were identified.

.5 High Risk Significant, High Dose Rate HRA and Very High Radiation Area (VHRA) Controls

a. Inspection Scope

The inspectors held discussions with the Radiation Protection Manager concerning high dose rate-HRA and VHRA controls and procedures, including any procedural changes that had occurred since the last inspection, in order to verify that procedure modifications did not substantially reduce the effectiveness and level of worker protection.

The inspectors discussed with RP supervisors the controls that were in place for special areas that had the potential to become VHRAs during certain plant operations, to determine if these plant operations required communication beforehand with the RP group, so as to allow corresponding timely actions to properly post and control the radiation hazards.

The inspectors also conducted plant walkdowns to verify the posting and locking of all accessible entrances to high dose rate-HRAs and VHRAs. These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.6 Radiation Worker Performance and Radiation Protection Technician Proficiency

a. Inspection Scope

The inspectors reviewed condition reports generated since the beginning of calendar year 2002 which identified that the cause of the event was related to radiation worker errors or radiation protection technician errors to determine if there was a trend due to a similar cause, and to determine if this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. These problems, along with planned and accomplished corrective actions, were discussed with Radiation Protection management.

These reviews represented two inspection samples; one sample for the reviews related to radiation worker errors, and one sample for the reviews related to radiation protection technician errors.

b. Findings

No findings of significance were identified.

2OS2 As Low As Is Reasonably Achievable (ALARA) Planning And Controls (71121.02)

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed plant collective exposure history, current exposure trends, ongoing and planned activities in order to assess current performance and exposure challenges. This included a determination of site specific trends in collective exposures and source-term measurements. These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Radiological Work Planning.

a. Inspection Scope

The inspectors compared the results achieved including dose rate reductions and person-rem used with the intended dose established in the licensee's ALARA planning for several work activities (and associated RWPs) conducted during the most recent refueling outage, including:

- Drywell In-service Inspections, including welds and preps, snubbers, and bioshield doors (RWP 03-1112);
- E11F050A and E11F050B Valves Hardseat Modification/Softseat Replacement (RWP 03-1127);
- Drywell Installation and Removal of Scaffold, Power, and Lights (RWP 03-1105);
- Drywell Shielding Installation/Removal (RWP 03-1103);
- Reactor Building Level 5 Refueling Activities (RWP 03-1251);
- Routine Maintenance/Instrument and Calibration Drywell Activities (RWP 03-1107); and
- Drywell Steam Relief Valve Repairs (RWP 03-1120).

The licensee's post-job (work activity) reviews were evaluated to understand the reasons for inconsistencies between intended and actual work activity doses and to verify that identified problems were entered into the licensee's corrective action program. These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

.3 Source-Term Reduction and Control

a. Inspection Scope

The inspectors reviewed licensee records to determine the historical trends and current status of tracked plant source terms, and to determine if the licensee had developed contingency plans for potential changes in the source term due to changes in plant fuel performance issues or changes in plant primary chemistry.

The inspectors further reviewed these documents to verify that the licensee had developed an understanding of the plant source-term, that this included knowledge of input mechanisms to reduce the source term, and that the licensee had a source-term control strategy in place (which included a cobalt reduction/maintenance strategy which was designed to minimize the source-term external to the core). These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

.4 Problem Identification and Resolution

The inspectors reviewed the licensee's self-assessments, audits, and Special Reports, as available, related to the ALARA program since the last inspection to determine if the licensee's overall audit program's scope and frequency for all applicable areas under the Occupational Cornerstone met the requirements of 10 CFR 20.1101(c).

The licensee's corrective action program was also reviewed to determine if repetitive deficiencies and/or significant individual deficiencies in problem identification and resolution, relative to the ALARA program, had been addressed. These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment (71121.03)

Respiratory Protective Equipment

a. Inspection Scope

The inspectors reviewed the licensee's respiratory protection procedure and discussed their implementation relative to the requirements of 10 CFR 20.1703(f) for standby rescue persons whenever one-piece atmosphere supplying suits, or any combination of respiratory protection and personnel protective equipment were used from which the wearer may have difficulty extricating himself. Specifically, the inspectors reviewed the licensee's work planning process and implementing practices, and interviewed RP staff regarding the following aspects of 10 CFR 20.1703: (1) designation of an adequate number of standby rescue workers and their training/instruction; (2) presence of equipment staged at the work site for the safety of the rescuer and for extrication of the respiratory equipment user; (3) practices for continuous communication between standby rescuer(s) and the respiratory protection user(s); and (4) provisions for immediate availability of the standby rescuer.

b. Findings

No findings of significance were identified.

**1. OTHER ACTIVITIES (OA)**

4OA1 Performance Indicator Verification (71151)

.1 Reactor Safety Strategic Area

a. Inspection Scope

The inspectors sampled the licensees submittals for performance indicators (PIs) for the period listed below. The inspectors used PI definitions and guidance contained in Revision 2 of Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," to verify the accuracy of the PI data. The following six PIs were reviewed:

- Safety System Unavailability, Emergency AC Power from January 1, 2002, through December 31, 2002;

- Residual Heat Removal System Unavailability from January 1, 2002, through December 31, 2002;
- Reactor Coolant System Leakage from January 1, 2002, through December 31, 2002;
- Alert and Notification System from July 2002 to June 2003;
- Emergency Response Organization Drill Participation from July 2002 to June 2003; and
- Drill and Exercise Performance from July 2002 to June 2003.

The inspectors reviewed selected applicable conditions and data from logs, licensee event reports and CRs written between the dates indicated for each PI area specified above. The inspectors independently re-performed calculations where applicable. The inspectors compared that information to the information required for each PI definition in the guideline to ensure that the licensee reported the data correctly.

b. Findings

No findings of significance were identified.

.2 Radiation Safety Strategic Area

a. Inspection Scope

The inspectors sampled the licensee's submittals for the PI and period listed below. The inspectors used PI definitions and guidance contained in Revision 2 of Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," to verify the accuracy of the PI data. The following PI was reviewed:

- Occupational Exposure Control Effectiveness

Specifically, the inspectors reviewed licensee Occupational Exposure Control Effectiveness PI assessment documentation for the period of the 2$^{nd}$ Quarter of 2002 through the 3$^{rd}$ Quarter of 2003. Since no reportable issues were identified by the licensee for this time period, the inspectors compared the licensee's data with the corrective action program database and radiologically restricted area electronic dosimetry transaction records to verify that there were no unaccounted for occurrences in the PI as defined by Revision 2 of Nuclear Energy Institute Document 99-02.

b. Findings

No findings of significance were identified.

40A2 Identification and Resolution of Problems (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

As discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's corrective action system at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Minor issues entered into the licensee's corrective action system as a result of inspectors' observations are included in the list of documents reviewed which are attached to this report.

b. Findings

No findings of significance were identified.

.2 Corrective Actions Involving Calculations and Engineering Functional Analyses

a. Prioritization and Evaluation of Issues

(1) Inspection Scope

The inspectors reviewed the engineering functional analyses and evaluations associated with corrective actions documented in CARDs 03-18712, 03-20243, and 03-12097. Specifically, the inspectors reviewed these corrective actions to ensure that all assumptions included current design basis information and appropriate equations, the engineering evaluations were both thorough and accurate, and that no immediate safety concerns existed.

(2) Issues

CARD 03-18712

On September 30, 2003, Division 2 control center HVAC (CCHVAC) was shifted to Division 1 to support a secondary containment surveillance, but the Division 1 chiller failed to start. The chiller was reset and operators unsuccessfully attempted a second start. By this time, Division 2 CCHVAC failed to start because the positioner for the fan blade would not move causing a protective trip of that system. Operations personnel entered Technical Specification 3.0.3 due to two inoperable divisions of CCHVAC and a plant downpower was started to comply with the specification. About 2 hours after the failures with the reactor at 84 percent power, the Division 1 chiller was successfully started and loaded. The licensee determined that air in the oil sensing line prevented a start permissive from actuating. (Air in the oil sensing line was a repeat issue as documented in NRC Inspection Report 05000341/2003009). Division 1 CCHVAC was declared operable and the power decrease was stopped.

In response to this event, a plant support engineer was requested to develop an EFA to justify operability of the Division 1 CCHVAC chiller. Chiller operability was partially predicated on the heatup rate of the control center, which was determined under another EFA written in 1999 for CARD 99-16680. This EFA, which used winter outside temperatures as an assumption, noted that the most limiting control center temperature was in the relay room at 104 degrees F, which would be reached within 31 hours. The engineer concluded in the 2003 EFA that the time to start the Division 1 CCHVAC was about 1 hour and was considered conservative when compared to the time calculated in CARD 99-16680.

The inspectors reviewed the 1999 and the 2003 EFAs and the applicable sections of the UFSAR. UFSAR Section 9.4.1.1 stated that two fan coil cooling units are located in the mechanical equipment room to dissipate the total heat load in the mechanical equipment room during an emergency. Chilled water is supplied to these units from the control center chillers. The engineer who developed both the 1999 and 2003 EFAs missed the fact that the CCHVAC chillers provided cooling in the mechanical equipment room; therefore, the heat load in the mechanical equipment room was missed. A further review of the 1999 EFA was performed based on this missed information and it was identified that the potential impact of coupling each of the control center compartments continued forced circulation on the overall heatup of all the rooms was not addressed. Finally, the only ventilation credited in the 1999 EFA was the introduction of cool outside air via the operation of the control room emergency filtration (CREF) system. In crediting the introduction of this air, the 1999 EFA took no account of the fact that the charcoal heaters in the CREF charcoal beds that maintain the makeup air provided an additional heat load.

The 2003 EFA was deficient because: 1) it cited the non-conservative analysis in the 1999 EFA as a basis for the allowable time for operator action and 2) the engineer failed to recognize that it was improper to use the 1999 EFA since the 1999 EFA was performed assuming winter conditions and not for summer weather conditions, as the loss of both CCHVAC divisions occurred in September.

Although a reevaluation was performed using summer air temperatures vice winter air temperatures, mechanical equipment room heat loads and the heat load from the charcoal bed heaters, the conclusion of the 2003 EFA remained unchanged. However the time to reach the maximum temperature of 104 degrees F in the relay room was reduced from 31 hours to 20 hours.

<u>CARD 03-20243</u>

During Forced Outage 03-01, a pressure seal leak (body to bonnet) was identified on the G3300F120 check valve. CARD 03-20243 was written to document the condition. Corrective Action Number 03-20243-01 required an evaluation of the leakage be prepared to establish a monitoring plan per Corrective Action Number 03-20243-02. This valve is a boundary valve located between the Reactor Water Cleanup discharge and the feedwater line and was leaking approximately 10 drops per minute.

An evaluation established an administrative leakage limit to assure that no limiting condition for operation would be exceeded. A leakage limit, based on air, of 14.5 scfh at

56.5 psig was identified as conservative as this was the value obtained during the last local leak rate test (LLRT) of this valve. Engineers used this value and the Darcy formula listed in the Crane "Flow of Fluids through Valves, Fittings and Pipe," handbook to determine an orifice size. From this orifice size, an equivalent water leakage of 771ml/min was determined assuming a reactor pressure of 1045 psig. The calculation was checked during that night and approved. The individuals who verified the accuracy of the evaluation had no more experience in fluid dynamics than the original preparers.

The next day, the inspectors reviewed the calculation using the Crane handbook as a guide. The inspectors noted that the Darcy formula was a general equation for compressible fluids, such as air, that had a restriction on its application based on pressure drop. For a pressure drop greater than 40 percent, the density and velocity of the fluid changes appreciably and the condition could be considered a "long pipe." Therefore the complete isothermal equation must be used, which was a completely different equation from the Darcy formula. The inspectors performed a rough calculation and determined that the pressure drop for this condition would have exceeded 40 percent. Based on the appearance that the licensee used the incorrect formula, the inspectors challenged the validity of the calculation and as such, the established administrative limit of 771 ml/min. In response to the inspectors' questions, additional personnel with comprehensive understanding of fluid dynamics reviewed the calculations. Based on this subsequent review, the leakage limit was revised to 249.8 ml/min.

<u>CARD 03-12097</u>

The inspectors reviewed CARD 03-12097 which was written to document the low EECW flow to the Division 2 RHR mechanical seal coolers. The inspectors reviewed the operability evaluation and determined that the evaluation was deficient in several ways.

The operability evaluation referenced an engineering review that determined that "the loss of the seals would not prevent the operation of the pump or the shutdown cooling function itself." This plant support engineering review only discussed the effects that a leaking mechanical seal would have on the ability of the pump shaft to continue turning. The review concluded that the pump's shaft would not "bind or freeze up preventing the pump from rotating." No written evaluation was performed to analyze the failure modes and effects as part of this evaluation. Specifically, the inspectors could find no evidence that an analysis of the consequences of the maximum credible seal leak was performed from an environmental qualification, flooding, or radiological safety standpoint.

The operability evaluation further stated that "only one residual heat removal pump is required in the shutdown cooling mode; hence, a backup would be available." The inspectors determined that the licensee inappropriately relied on redundancy to help justify operability.

As a result of the inspectors' concerns, the licensee performed a more in-depth analysis of the issue including contracting General Electric Nuclear Engineering to study the issue. The licensee concluded that a RHR seal failure would result in minor seal leakage that would be easily identified and corrected.

Although the outcome of each issue (CARDs 03-18712, 03-20243, and 03-12097) was relatively minor, this demonstrated that engineering functional analyses and evaluations developed as corrective actions for these CARDs lacked thoroughness and improper use of formulas.

.3 Corrective Actions Involving Maintenance Rule Monitoring

a. Effectiveness of Corrective Actions

(1) Inspection Scope

The inspectors reviewed multiple related condition reports to determine if the corrective actions implemented were effective in preventing a recurrence of a similar issue.

(2) Issues

The inspectors reviewed four NRC-identified instances of inadequate maintenance rule monitoring as described in CARDs 01-20897, 02-11760, 03-18539, and 03-23263. The inspectors determined that although an adverse trend in maintenance rule monitoring existed, none of the issues discovered were more than minor.

The following issues were NRC-identified:

- The emergency hotwell supply pump was tagged out of service on November 27, 2000, but that the unavailable hours were not captured as required (CARD 01-20897);
- The failure of the HPCI room floor drain isolation valve on December 7, 2001, did not receive a functional failure analysis (CARD 02-11760);
- Reactor Water Cleanup risk significant isolation valve out of service hours were not tracked since April 2002 (CARD 03-18539); and
- On November 28, 2001, a half SCRAM received while lining up a reactor feedwater pump was incorrectly evaluated as not being maintenance preventable (CARD 03-23263).

In the first three examples above, none of the systems would have exceeded their associated performance criteria had maintenance rule monitoring been more effective. Regarding the improper half SCRAM evaluation, the licensee determined that the system did exceed its maintenance rule performance criteria of no more than three unplanned maintenance preventable half SCRAMS per two cycles. Because the licensee did not properly capture the performance criteria, no a(1) evaluation by the licensee's expert panel was conducted. The inspectors concluded that this issue was minor because it did not represent an actual degradation in the reliability or ability of the reactor protection system to initiate and complete a reactor SCRAM.

The inspectors, as well as the licensee, determined that these issues represented an adverse trend in maintenance rule monitoring. The licensee planned to develop corrective actions to strengthen their maintenance rule program monitoring.

4OA3 Event Followup (71153)

Review of Licensee Event Reports

a. Inspection Scope

The inspectors performed an onsite review of records to evaluate the root cause and corrective actions for the LERs discussed in the "Findings" section below. The inspectors evaluated the timeliness, completeness, and adequacy of the root cause and corrective actions in accordance with the requirements of 10 CFR Part 50, Appendix B, as appropriate.

b. Findings

.1 (Closed) LER 50-341/03-001, "Loss of High Pressure Coolant Injection Safety Function Due to Closure of Steam Supply Valve"

This LER involved the unplanned steam supply isolation due to the failure of the outboard steam supply valve to fully close rendering High Pressure Coolant Injection (HPCI) inoperable. On July 13, 2003, during HPCI surveillance testing, the outboard steam supply isolation valve failed to fully close when the close pushbutton was depressed due to a random failure of the auxiliary seal-in contact for the valve closing circuit. The inboard steam supply isolation valve was closed to comply with Technical Specification 3.6.1.3, Primary Containment isolation valves. The inspectors verified that the corrective actions for CARD 03-10957, which had been written to document the event were completed and implemented. This LER is closed.

.2 (Closed) LER 50-341/03-002, "Automatic Reactor Shutdown Due to Electric Grid Disturbance and Loss of Offsite Power"

This LER involved the reactor shutdown that occurred as a result of the international blackout on August 14, 2003. At approximately 4:10 p.m., the plant experienced a complete Loss of Offsite Power (LOOP) with subsequent turbine trip and reactor SCRAM. The inspectors reviewed that the corrective actions for CARD 03-19948, which had been written to document the event were completed and implemented. The inspectors also reviewed the licensee's reporting of this event to ensure that all notifications were made as required. This LER is closed.

4OA4 Cross-Cutting Aspects of Findings

A finding described in Section 1R022.2 of this report had, as its primary cause, a human performance deficiency, in that, an Instrumentation and Controls Technician failed to perform adequate self-checking and improperly connected test equipment to the wrong terminals to perform the RCIC steam diaphragm pressure test. This caused a short circuit and an unexpected steam isolation of the RCIC system.

4OA5 Other

1. (Closed) URI 50-341/2003-008-01: Station Blackout Combustion Turbine Generator Failure to Start During Loss of Offsite Power Grid

Introduction. The inspectors identified a self-revealed Green finding regarding an improper modification process used in 1996 to install an inverter on station blackout CTG 11-1 that did not include updating the design basis central component (CECO) database of the appropriate low voltage inverter trip set point. The low voltage trip set point was set too high and prevented CTG 11-1 from starting on demand during the loss of offsite power event on August 14, 2003. The finding was not considered a violation of regulatory requirements. The finding represented a decrease in availability, reliability, and capability of the station blackout combustion turbine generator to respond to initiating events to prevent undesirable consequences (i.e., core damage).

Description. The inspectors documented in Inspection Report 05000341/2003008 that CTG 11-1 tripped during a demand start following the loss of the power grid on August 14, 2003, because the low voltage trip set point on the inverter was set too high.

Originally, a motor generator (MG) set on CTG 11-1 was used to provide 120VAC control power and power to the exciter. A decision was made in 1996 to replace the MG set with an inverter. No documentation justifying the reason for the change existed. The vendor provided an inverter with factory set low and high inverter voltage trip set points based on a system that has a battery bank of 60 cells. The low voltage set point is determined at 1.75 volts/cell thereby establishing a 105-volt low factory set point. However, CTG 11-1 has a 56-cell battery bank requiring the low voltage factory set point be changed to 98 volts. The inverter was installed without changing the set points. Because the modification process was not used the central component (CECO) information database that contains design basis information, particularly set point values, was not updated to reflect the correct set points.

The inverter had the incorrect low voltage set points for about 6 months until the set point card S2A-167 was changed. During this card replacement, personnel referenced the vendor manual and recognized the 1.75 volt/cell requirement, counted 56 cells on CTG 11-1 and correctly adjusted the low voltage trip set point to 98 volts, installed the card and placed CTG 11-1 in service.

On August 22, 2001, the card burned and was replaced. Since CECO did not contain the low voltage trip set points, the work package to replace the card did not include instructions to change the factory set low voltage trip set point of 105 volts to 98 volts. On August 14, 2003, when a loss of offsite power occurred, an initial attempt to start CTG 11-1 using DC power from the batteries failed because the large voltage drop during the start sequence had dropped below 105 volts and the inverter tripped.

Analysis. The inspectors determined that CTG 11-1 failed to start during the loss of offsite power event because the inverter low voltage trip set point was set too high. Installing the circuit card without properly adjusting the set point was the result of inadequate implementation of the modification process. The inspectors concluded that the finding was greater than minor in accordance with IMC 0612, “Power Reactor

Inspection Reports," Appendix B, "Issue Disposition Screening," issued on June 20, 2003. The inspectors determined that the finding was more than minor because it affected the mitigating systems cornerstone objective to ensure availability, reliability, and capability of the CTG 11-1 system that responds to initiating events to prevent undesirable consequences (i.e., core damage).

The inspectors determined that the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," because the finding was associated with the failure of the station blackout combustion turbine generator to start when called upon during the station blackout event of August 14, 2003. Using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors answered "no" to all five screening questions in the Phase 1 Screening Worksheet under the mitigating systems column. The inspectors concluded the issue was of very low safety significance.

Enforcement. The station blackout combustion turbine generator was a non-Technical Specification system and not required by 10 CFR Part 50, Appendix B; therefore, no violation of regulatory requirements occurred. This issue was considered a Green finding of very low safety significance **(FIN 05000341/2003010-03)**. The licensee entered improvement plans for CTG 11-1 into the corrective action program under CARD 02-24766.

2. (Closed) Unresolved Item (05000341/2003007-01(DRS)): This unresolved item was issued for RHR heat exchangers at Fermi that could be a potential unmonitored release path for radiation during a design basis accident (DBA). The RHR heat exchangers were not eddy current tested since plant construction. Therefore, the condition of the heat exchanger tubes, such as thinning or leakage, was unknown. Although there was no known evidence of tube leakage or degradation, the impact of possible leakage (given the age of the heat exchangers and lack of licensee inspections) had not been analyzed. Specifically, tube leakage during a DBA would allow radioactive suppression pool water to enter the RHRSW System and be pumped directly outside into the environment. It was not clear whether the current methods of monitoring and sampling would provide a reliable indication of actual radiological releases or provide warning of off-normal conditions under these circumstances in time to terminate a release prior to exceeding regulatory limits. However, the inspectors did not identify any NRC requirements or licensee commitments to do visual or eddy current testing of these heat exchangers. As a result of questions from the inspectors, the licensee evaluated their current approach to RHR heat exchanger tube degradation monitoring and decided to eddy current test the RHR heat exchangers as soon as they could fit the work into an outage. No violation of NRC requirements occurred. This item is closed.

## 4OA6 Meetings

### .1 Exit Meeting

The inspectors presented the inspection results to Mr. O'Connor and other members of licensee management at the conclusion of the inspection on January 9, 2004. The inspectors asked the licensee whether any material examined during the inspection should be considered proprietary.  One proprietary document (that analyzed the effects of low cooling flow to the RHR pump seal coolers) was identified which was returned to the licensee.

.2 Interim Exit Meetings

Interim exits were conducted for:

- Heat Sink Inspection with Mr. W. O'Connor on October 31, 2003;
- Emergency Preparedness inspection with Mr. W. O'Connor on October 31, 2003;
- Occupational Radiation Safety access control and ALARA programs inspection with Mr. D. Cobb on November 7, 2003; and
- Biennial Operator Requalification Program Inspection with Mr. W. O'Connor on November 21, 2003.

4OA7 Licensee-Identified Violations

No findings of significance were identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

## SUPPLEMENTAL INFORMATION

## KEY POINTS OF CONTACT

Licensee

W. O'Connor, Jr., Vice President Nuclear Generation
D. Cobb, Plant Manager
H. Higgins, Manager, Radiation Protection
J. Korte, Nuclear Security Manager
R. Libra, Nuclear Engineering Manager
K. Morris, Emergency Preparedness Supervisor
J. Myers, Nuclear Quality Assurance Manager
N. Peterson, Nuclear Licensing Manager
M. Philippon, Operations Manager
L. Sanders, Nuclear Training Manager
P. Smith, Nuclear Fuels/PSA Manager
S. Stasek, Nuclear Assessment Director

NRC

M. Ring, Chief, Division of Reactor Projects, Branch 1

## LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

| | | |
|---|---|---|
| 05000341/2003010-01 | NCV | 10 CFR 50, Appendix B, Criterion V violation for failure to follow procedures resulting in a RCIC steam isolation. |
| 05000341/2003010-02 | NCV | 10 CFR 50.54(q) Violation for decreasing the effectiveness of the E-Plan by changing EALs that address toxic gases without prior NRC approval. |
| 05000341/2003010-03 | FIN | Inadequate implementation of the modification process prevented CTG 11-1 from starting during the August 14, 2003 LOOP. |

Closed

| | | |
|---|---|---|
| 05000341/2003010-01 | NCV | 10 CFR 50, Appendix B, Criterion V violation for failure to follow procedures resulting in a RCIC steam isolation. |
| 05000341/2003010-02 | NCV | 10 CFR 50.54(q) Violation for decreasing the effectiveness of the E-Plan by changing EALs that address toxic gases without prior NRC approval. |

| | | |
|---|---|---|
| 05000341/2003007-01 | URI | Possible failure to provide a program with controls to measure and minimize leakage of highly radioactive fluids outside containment during a serious transient or accident. |
| 05000341/2003-001 | LER | Loss of HPCI safety function due to closure of steam supply valve. |
| 05000341/2003-002 | LER | Automatic reactor shutdown due to electric grid disturbance and LOOP. |
| 05000341/2003008-01 | URI | Station blackout combustion turbine generator (CTG)failure to start during loss of electrical grid. |
| 05000341/2003010-03 | FIN | Inadequate implementation of the modification process prevented CTG 11-1 from starting during the August 14, 2003 LOOP. |

Discussed

None.

## LIST OF DOCUMENTS REVIEWED

1R01 Adverse Weather

System operating procedure 23.416, Rev. 9; General service water pump house heating and ventilation

ARP 11D50, Rev. 11; CTG 11 unit 1 trouble

CARD 03-13516; Unable to produce effective warm/cold weather monthly report; dated October 13, 2003

Performance evaluation procedure 27.000.04, Rev. 26; Freeze protection lineup verification

Safety tagging record number 2003-001459

Drawing 6D-2522-38, Rev. C; Interconnection diagram 480 VAC MCC 72K-2A BOP general service water pump house

UFSAR 9.4.9; General service water pump house heating and ventilation system

UFSAR figure 9.4-12; General service pump house ventilation system flow diagram

UFSAR table 9.4-14; General service water pump house heating and ventilation system components descriptions

System operating procedure 23.104, Rev. 63; Condensate storage and transfer system

System operating procedure 23.501.01, Rev. 37; Fire water suppression system

System operating procedure 23.324, Rev. 48; Supervisory control - 120 kV switchyard and CTG11 generators

1R04 Equipment Alignment

Functional Operating Sketch 6M721-5706-3, “RHR Service Water Make Up Decant and Overflow Systems,” Revision U

Functional Operating Sketch 6M721-5706-2, “RHR Division 1,” Revision W

Functional Operating Sketch 6M721-5734, “EDG System,” Revision AK

Procedure 23.307, “EDG 11 Valve Lineup,” dated 8/28/03

One Line Diagram 6SD721F-0024, “480V AC 30, 12/240 AC 10 and 125V DC Buses CTG 11,” Revision H

Procedure 23.309, “260/130V DC Electrical System (ESF and BOP),” Revision 42

Procedure 23.310, “48/24/V DC Electrical System,” Revision 22

WR 000Z013869, “Division 1 RHRSW Pump C Minimum Flow Valve,” dated 1/4/02

Procedure 23.324, “ Supervisory control - 120kV switchyard & CTG11,” Revision 48

Temporary Modification 03-0018, “CTG 2, 3 or 4 can be used as the Dedicated Fermi 2 Alternate AC Power Supply to Substitute for the Black Start Unit CTG 11-1,” Revision B

CARD 00-13077; Discrepancies on MCC compartment terminals for MCC 2PA-1 Pos 5A and 5B on schematic I-2231-06; dated March 27, 2000

CARD 00-15252; Supports need to be installed for RCIC overspeed trip motor; dated May 25, 2000

CARD 00-15470; Found existing stem nut with inadequate counterbores; dated May 6, 2000

CARD 00-16781; Re-route RCIC pump seal water piping; dated September 7, 2000

CARD 00-18882; RCIC SOP valve lineup correction 23.206; dated August 2, 2000

CARD 00-24359; Drawing configuration incorrect; dated November 29, 2000

CARD 01-13093; Errors on drawing SD-2530-14; dated April 30, 2001

CARD 01-19332; RCIC cyclone separators possibly piped wrong; dated October 24, 2001

CARD 02-20966; Error on DWG I-2045-56; dated December 4, 2002

CARD 03-15085; Scaffold builders removed handwheels on RCIC system to build scaffold; dated April 15, 2003

CARD 98-10344; Errors made during revision K update on drawing I-2791-01; dated January 27, 1998

CARD 99-17017; Missing thrust washers on RCIC turbine governor valve control linkage; dated September 8, 1999

CARD 00-12510; RCIC turbine overspeed trip limit switch actuator is loose on its shaft; dated April 30, 2000

CARD 99-16916; New MCC positions; dated September 10, 1999

CARD 99-18377; Incorrect valve opening spring found installed on RCIC turbine governor valve; dated December 8, 1999

ST-OP-315-0043-001, Rev. 10; Operations training - RCIC

Drawing M-5709-1, Rev. AC; RCIC system sketch functional operating sketch

Drawing M-2044, Rev. AX; Diagram RCIC system

Drawing M-5709-2, Rev. F; RCIC turbine lube oil/control oil functional operating sketch

System operating procedure 23.206, Rev. 75; RCIC system

1R05 Fire Protection

MOP 10, Revision 2; Operations Conduct Manual Chapter 10 - Fire Brigade.

Fermi Fire Protection Pre-Plans.

1R06 Flood Protection

WR B447030100, "N2100-F608 S RFP Discharge Line ISO Valve," dated 7/15/02

UFSAR 3.4.4.4, "Internal Flood Protection"

WR 000Z014483, "T4500-F601 HPCI Room Floor & Equipment Drain ISO Valve," dated 1/22/02

Procedure 27.702.01, "Reactor Building Sump Crosstie Flood Control Valve Test," dated 2/21/02

WR J6900020100, "Recalibrate Sump G1101D073 Level Loop," dated 12/23/02

UFSAR Figure 3.6-13, "Selective Room Locations Reactor Building Subbasement Elevation 540.0ft"

UFSAR Figure 3.6-36, "Reactor/Auxiliary Building First Floor," Revision 3

UFSAR Figure 9.3-6, "Floor Drains in Auxiliary and Reactor Buildings," Revision 7

USFAR Table 3.6-6, "Flow and Events Postulated for Feedwater Break"

CARD 03-21570, "Recorder Display Garbled," dated 10/5/03

CARD 01-19553, "Valve Failed to Close During Performance of 27.702.01," dated 12/7/01

CARD 02-12020, "Mechanical Binding in Valve Causing Premature Closing Torque Switch Trips," dated 3/7/02

Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants," Revision 1

Procedure ARP 2D105, “Reactor Building Corner Room/HPCI Room Flood Level,” Revision 12

Diagram 6M721-2224, “Floor Drains All Floors Auxiliary and Reactor Buildings,” Revision J

Diagram 7M721-2218, “Floor and Equipment Drains, Sub-Basement Plan - Reactor Building - Unit 2,” Revision W

<u>1R07 Biennial Review of Heat Sink Performance</u>

EFA-P44-02-005; Emergency Equipment Service Water (EESW) Operability With Additional Drywell Heat Load Due to Loss of a Division of Drywell Cooling Without Subsequent High Drywell Pressure Signal; dated September 19, 2002

EFA-P44-03-007; Low Predicted Deliverable Flows for Emergency Equipment Cooling Water (EECW); dated November 25, 2002

EFA-P44-03-014; EECW Flow Rates to Thermal Recombiner and Battery Charger Room Coolers Found Low Out of Bands; dated November 25, 2002

Generic Letter 89-13; Service Water System Problems Affecting Safety-Related Equipment; dated July 18, 1989

Generic Letter 89-13; Service Water System Problems Affecting Safety-Related Equipment, Supplement 1; dated April 4, 1990

DER 89-0880; Generic Letter 89-13: Service Water System Problems Affecting Safety-Related Equipment; dated August 2, 1989

Response to Nuclear Regulatory Commission Generic Letter 89-13 for Fermi 2 Nuclear Plant Emergency Equipment Cooling Water System Single Failure Capability; dated April 9, 1990

Information Notice 86-60; Unanalyzed Post-Lost of Coolant Accident (Post-LOCA) Release Paths; dated July 28, 1986

TMBL-96-0006; General Design Criteria 64 Commitment Met for post-LOCA Conditions; dated February 14, 1996

Memorandum; Response to NRC Generic Letter 89-13; dated January 26, 1990

Memorandum EF2-49082; IE Bulletin 80-10: Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment; dated May14, 1980

Memorandum EF2-72701; Bulletin 80-10; dated July 24, 1984

Memorandum EF2-56,325; Potential Contamination of Non-radioactive Systems; dated

January 20, 1982

Memorandum; Follow-up Action and Response Regarding the Three Mile Island Unit 2 Accident; dated February 29, 1980

Memorandum; Fermi 2 Individual Plant Examination Program for Severe Accident Vulnerability - Generic Letter 88-20; dated October 25, 1989

Offsite Dose Calculation Manual; Revision 14

MHL96052; Memorandum: Residual Heat Removal (RHR) Pump Seal Cooling; dated September 18, 1996

Inservice Test Data for EESW Pumps 1996 through 2003; dated October 29, 2003

List of Inservice Test Limits for EECW and EESW Pumps; dated November 30, 2003

Fermi 2 Program Health Report; Heat Exchanger Program; dated 2002

TG25010930; Perform 47.205.002 RHR Division 2 Heat Exchanger Performance Test; dated April 22, 2003

AJ93030822; Perform SOE 03-12 Division 2 EECW Flow Balance; dated August 23, 2003

AJ93030911; Division 2 EECW Throttled Valve Flow Verification; dated August 23, 2003

47.207.01; Emergency Equipment Cooling Water Division Heat Exchanger Performance Test; Revision 31

Chemistry Gamma Spectroscopy Analysis Report for Reactor, dated March 29, 2003

File 133907; Chemistry Gamma Spectroscopy Analysis Report; dated April 29, 2003

Reactor Tritium Activity Calculation for April 3, 2003 and May 9, 2003 and May 23, 2003; dated June 2, 2003

SPC-13682; RHR Service Water Radiation Monitor Set-point Increase; dated December 18, 1992

List of Condition Assessment Resolution Documents (CARDs) Initiated Since July 1, 2001 for Heat Exchangers; dated August 19, 2003

CARD 01-02217; Calibration of Instruments Used to Perform Shutdown Cooling System Surveillances; dated November 17, 2001

CARD 01-13222; Current “Calibrated” Division II EECW Hydraulic Model May Not Accurately Predict Division II EECW Flows; dated July 23, 2001

CARD 01-13239; Logarithmic Mean Temperature Difference (LMTD) Correction Factor of 1.0 Used in RHR Heat Exchanger Test Analysis; dated August 2, 2001

CARD 01-13240; RHR Heat Exchanger Test Acceptance Criteria; dated August 2, 2001

CARD 01-13241; RHR Heat Exchanger Design Fouling Less Than the Fouling Allowed in the RHR Heat Exchanger Performance Test; dated August 2, 2001

CARD 01-16108; Evaluate Closed Function of Residual Heat Removal Service Water (RHRSW) Heat Exchanger Discharge Check Valves; dated July 9, 2001

CARD 01-19530; Potential Excessive Flow Through Division 1 RHR Heat Exchanger; dated November 8, 2001

CARD 01-21956; Flows to EECW Throttled Loads Outside Required Bands; dated December 11, 2001

CARD 02-12443; Recommend Using Two RHRSW Pumps During Dedicated Shutdown Scenario; dated September 5, 2002

CARD 02-19402; Discrepancy in RHR Heat Exchanger TEMA Data Sheet Shell Side Velocity; dated October 7, 2002

CARD 03-11884; Potential Release Path Not Accounted for in Off-site Dose Calculation Manual (ODCM); dated June 5, 2003 (NRC Identified)

CARD 03-11894; No Retrievable Documentation of the Design Temperatures and Cooling Water Requirements for RHR and Core Spray Pump Seals; dated June 6, 2003 (NRC Identified)

CARD 03-11939; OE 03-16259 - Residual Heat Removal Heat Exchanger Eddy Current Testing Reveals Pitting; dated May 27, 2003

CARD 03-12073; Nuclear Energy and Technology Center Calculation DET-07-007, Revision 3, Prediction of Post Accident Standby Gas Treatment Room Temperature (85 F Max) Inconsistent With EECW Cooling Water Maximum Temperature of 95 F; dated May 9, 2003

CARD 03-12077; Check Valve Function Credited in Medium Energy Line Break Calculations But Not Tested; June 5, 2003 (NRC Identified)

CARD 03-12086; Potential Inadequate 50.59 for EDP-29805 (EESW Heat Exchanger Replacement); dated May 29, 2003

CARD 03-12097; EECW Flow to E1102C002(D) Seal Coolers Measured Under Safety Operability Evaluation (SOE) 03-01 Significantly Reduced From Flows Observed in SOE 95-11; dated March 24, 2003

CARD 03-14451; Root Cause Report - RHR Reservoir Service Water Pumps’ Bolt and Column Degradation; dated April 30, 2003

CARD 03-18031; Evaluate RHRSW Chemistry Sampling for Potential Enhancements for Monitoring Heat Exchanger Tube Leakage; dated May 29, 2003 (NRC Identified)

CARD 03 18033; Evaluate Revising Procedure 23.626 Process Liquid Radiation Monitoring to Enhance RHR Heat Exchanger Leak Detection; dated May 29, 2003 (NRC Identified)

CARD 03-18733; AFC Calculation DC-5540 Volume 1 “EECW Heat Exchanger Performance Requirements” is Supposed to be Superceded by DC-5806; dated October 29, 2003 (NRC Identified)

DBD P45-00; Emergency Equipment Service Water System Design Basis Document; dated November 16, 1998

DBD-E11-XX; RHR Service Water System Design Basis Document; dated December 3, 1990

E11-00; Design Basis Document RHR System; dated July 14, 2000

D11-00; Design Basis Document for Process Radiation Monitoring System; Revision A

DC-5805; EESW Design Basis Requirements; dated December 17, 1999

DC-559; Volume of Reservoir - RHR Complex; dated August 26, 1992

DC-5806; EEC Design Basis Requirements Calculation; Revision 0

DC-5806, Volume1, Appendix B; Summary of 11 EECW System Cooler Heat Loads from DC-5589; Revision 0

EDP-29805; Replacement of EECW Heat Exchangers; dated October 26, 1999

Residual Heat Removal Exchanger Specification Sheet; Revision 0

Project Specification 3071-088; Emergency Equipment Cooling Water System Heat Exchangers - Reactor Building; dated April 1, 1985

SE 99-0009; EEC Heat Exchanger Replacement; Revision 0

<u>1R11 Licensed Operator Requal</u>

Evaluation Scenario SS-OP-904-0020, “SDV Vent & Drain Failure, Loss of Air, SRV Open, LOCA,” dated 9/2/03

Evaluation Scenario SS-OP-904-0017, “Fire Impairment, Loss of RBCCW and Vacuum, ATWS,” dated 9/2/03

Six Licensed Operator Medical Records

List of Design Changes Completed During 2002 and 2003

Fermi 2 Nuclear Power Plant; Plant Issue Matrix; dated November 12, 2003

Fermi Written Examination Sample Plan Describing Which Training Areas Were Tested and Which Crews Were Seeing Which Questions

Fermi Job Performance Measure Sample Plan and Completion Matrix Documenting Which Operators Had Seen Which Job Performance Measures For the Past Two Calendar Years

Fermi Dynamic Simulator Sample Plan

Simulator Open Work Request Report; October 20, 2003

Completed Simulator Work Request Report; October 20, 2003

Malfunction Event #018; Main Generator Trip; Performed October 12, 2002

Malfunction Event #034; Condenser Air Leak Performed October 21, 2002

Surveillance Test 24.137.01; Main Steam Line Isolation Channel Functional Test Performed; October 8, 2002

Letter From W.S. Orser to Director, NRR; RE: Certification of the Fermi 2 Upgraded Simulator Facility; dated December 30, 1992

Nuclear Training Work Instruction 1.15, “Simulator Maintenance;” Revision 4

Corrective Action Record Document (CARD) 0122208; “Level 8 Trip While Unisolating the North Reactor Feed Pump”

CARD 0219225; “Emergency Operating Procedure Interaction With Post-Fire Shutdown Using Abnormal Operating Procedure 20.000.18"

CARD 0317206; “Loss of Shutdown Cooling Due To Operator Error”

NRC Inspection Report 50-341/01-13(DRP)

NRC Inspection Report 50-341/03-06

NRC Inspection Report 50-341/01-17(DRP)

NRC Inspection Report 50-341/02-08(DRS)

NRC Inspection Report 50-341/02-07

Evaluation Scenario; SS-OP-904-0010; “Scram Discharge Volume Vent & Drain Failure, Loss of Air, Safety Relief Valve Open; Loss of Coolant Accident;” Revision 0

Evaluation Scenario; SS-OP-904-0013; “Standby Liquid Control & Control Room

Ventilation Failures, Loss of Turbine Building Closed Cooling Water, Anticipated Transient Without Scram, Reactor Pressure Vessel Flooding;" Revision 0

Job Performance Measure (JPM); "JP-OP-3006-500; Drywell Vent In Accordance With 29ESP07 (Alternate Path); Revision 1

JPM; JP-OP-315-0165-001; "Parallel An EDG [Emergency Diesel Generator] From The Control Room;" Revision 1

JPM; JP-OP-315-0104-010; "Recirc Pump Trip After Speed Increase;" Revision 0

Dynamic Simulator Crew Evaluation Forms For Scenarios SS-OP-0904-0010 and SS-OP-0904-0013; dated November 18, 2003

JPM Individual Evaluation Forms; dated November 19, 2003

CP-OP-202; Licensed Operator Requalification; Revision 16

CP-OP-232; Annual Requalification Examination; Revision 9

QP-OP-915; Licensed Operator Training and Qualification Program Description; Revision 4

Instruction 4; Conduct of Simulator Assessments & Evaluations; Revision 4

Simulator Summaries for 2002 Annual LOR Exam

ODE-8; Operations Department Expectation; Revision 2

03-02-OP; Self Assessment; 7/17/03 - NANT-03-0111

Licensed Operator Curriculum Review Committee (CRC) Meeting Minutes (1/30/03, 2/28, 6/20, 8/01, & 10/02)

LP-OP-202-0231; Cycle 3 Classroom Training; Revision 0

LP-OP-202-0221; Cycle 2 Classroom Training; Revision 0

Access Database Modification List; Change Tracking.mdb

ODE-8; Attachment 3 - Shift 1 (2, 3, 4, 5, and Off Shift) Active License Required Hours for 2003 (Quarters 1 - 3); Revision 0

<u>1R12 Maintenance Rule Implementation</u>

Selected control room operator logs from January 1, 2000, through November 24, 2003

CARD 00-15129; C4100F006 did not meet acceptance criteria during 24.139.03; dated May 2, 2000

CARD 01-20743; Check valve fails 47.000.13 acceptance criteria; dated November 8, 2001

CARD 01-20114; C4100F007 failed LLRT; dated November 5, 2001

CARD 01-20113; C4100F006 failed LLRT; dated November 4, 2001

Fermi maintenance rule conduct manual

CARD 03-16599; LLRT failure of the RWCU Div. 2 containment isolation MOV; dated April 5, 2003

C4100 system monthly report from November 1, 2000, through October 1, 2003

CARD 01-19394; SLC continuity loss (C41-F004B); dated October 8, 2001

CARD 01-20273; Evaluate PS 38555 and PS 38556: SLC system relief valves may lift during an ATWS; dated December 14, 2001

CARD 02-13799; Applicability of NRC info notice 2002-05 foreign material in SLC tank to Fermi 2; dated February 2, 2002

CARD 02-15634; Engineered protection required to prevent repeated damage; dated April 30, 2002

CARD 00-10851; As-left torque switch setting higher than CECO allowable maximum; dated April 14, 2000

CARD 03-11405; G3352F004 does not open after first open signal; dated April 15, 2003

CARD 03-11558; Higher actuator torque capability for DC MOVs; dated March 4, 2003

CARD 03-14577; Packing leak on valve G3352F220; dated April 30, 2003

CARD 03-16019; Degraded grease condition in MOV G3352F220; dated April 11, 2003

CARD 03-16599; LLRT failure of the RWCU Div. 2 containment isolation MOV; dated April 5, 2003

CARD 03-21457; G3352F220 failed to stroke full open; dated September 5, 2003

Maintenance rule scope determination for standby liquid control system (C4100); approved May 24, 1995

System health report for standby liquid control for 3rd and 4th quarters of 2002 and 1st and 2nd quarters of 2003

Design basis document no. C41-00, Rev. 0

Drawing I-2131-01, Rev. J; Schematic diagram standby liquid control pumps c4103c001A & B

Drawing M-5704, Rev. I; Standby liquid control system functional operating sketch

Drawing I-2265-03, Rev. P; Schematic diagram reactor water cln. up system isol. vlv G3352-F001, F004, F119, & F220 control circuits

Drawing M-2082, Rev. AA; Diagram standby liquid control system

1R13 Maintenance Risk Assessment and Emergent Work

NOED No. 01-3-001

NOED No. 99-6-007

Risk assessment memo for the week of December 1, 2003

Maintenance rule conduct manual, MMR12, Rev. 0

Downpower plan on December 5-6, 2003

Fermi 2 plan of the day; dated December 5, 2003

1R15 Operability Evaluations

EFA E41-03-001, Rev. 0; HPCI pump suction high point vent valve was found open and is missing it's packing gland follower nuts and bolts

Drawing M-5708-1, Rev. AH; High pressure coolant injection system functional operating sketch

Job ID 2100030708; Perform 24.202.01 Section 5.3, HPCI pump time response and operability test at 1025 psig

WR 000Z033745; Replace packing gland nuts/bolts found missing

ARP 2D51, Rev. 11; HPCI pump suction pressure high

CECO component information on E41N031

Vendor manual VMR1-3.1.2, Rev. A; Barksdale controls diaphragm pressure switches

EFA E41-02-007, Rev. 0; CARD 01-14752 states that the current PSA model does not credit NIAS as a required support for HPCI system availability

Design calculation DC-4931, Rev. E; Non interruptible control air system (NIAS) calculations

CARD 01-19531; Failed leakage/usage surveillance for Div. 2 NIAS; dated November 8, 2001

Job ID AG82030328; Perform 27.129.05 Div. 2 NIAS leakage/usage - compressor performance test

CARD 01-19481; Div. 2 NIAS failed acceptance criteria for section 5.2 (27.129.05) system usage/leakage; dated October 29, 2001

WR 000Z013524; High pressure coolant injection (HPCI) cooling water to turbine lube oil cooler E4100B002 pressure control valve

Job ID AG82011024; Perform 27.129.05 Div. 2 NIAS leakage/usage - compressor performance test

Job ID AG82011108; Perform 27.129.05 Div. 2 NIAS leakage/usage - compressor performance test

CARD 03-18712; Nonconservative analysis performed in 1999 EFA cited in support of currently active CCHVAC EFA (T41-03-016); dated October 6, 2003

EFA T41-03-016, Rev. 0

EFA T41-03-016, Rev. A

UFSAR 9.4.1; Control center air conditioning system

UFSAR 6.4.2.3; Air conditioning system

CARD 99-16680; During the winter months EECW temperature drops below 65 deg. F due to a mechanical stop in the EECW TCV. Analysis of components cooled by EECW have not been performed for temperature below the system temperature green band lower limit of 65 deg. F; dated September 3, 1999

<u>1R16 Operator Workarounds</u>

CARD 02-15674; N22K800A controller for N22F415A automatic signal failed low; dated June 9, 2002

Active operations challenges current as of November 20, 2003

DECO file no. NPOP-03-0043; Memo from Greg Strobel to Don Cobb & Mike Philippon; Aggregate assessment of operator work arounds; dated July 28,2003

DECO file no. TMSA-03-0059; Memo from Bob Slottke to Greg Strobel; Risk assessment of revised operator work arounds - July 2003; dated July 23, 2003

Open operator challenges for November 2003

CARD 02-13661; Feedwater heater level controls are challenging the operators; dated April 4, 2003

1R19 Post Maintenance Testing

CARD 03-21588, “Repack P4400F614 in RF10" dated 10/14/03

Inservice Test Program, “B2100, 2103, 2104, 21 Nuclear Boiler Valve Tables,” Revision 5

Inservice Test Program, “G1100 Radwaste System Valve Tables,” Revision 5

Inservice Test Program, “G3300 RWCU System Valve Tables,” Revision 5

Inservice Test Program, “G5100 Torus Water Management System Valve Tables,” Revision 5

Inservice Test Program, “N1100 Main Steam Supply System Valve Tables,” Revision 5

Inservice Test Program, “N2100 Feedwater System Valve Tables,” Revision 5

Inservice Test Program, “P1100 Condensate Storage System Valve Tables,” Revision 5

Inservice Test Program, “P3400 Post Accident Sampling System Valve Tables,“ Revision 5

Inservice Test Program, “P4200 RBCCW System Valve Tables,” Revision 5

Inservice Test Program, “P4400 EECW System Division 1 Valve Tables,” Revision 5

Inservice Test Program, “P4400 EECW System Division 2 Valve Tables,” Revision 5

Inservice Test Program, “P4500 EESW System Division 1 Valve Tables,” Revision 5

Inservice Test Program, “P5000 Station and Control Air System Valve Tables,” Revision 5

Inservice Test Program, “R3000 Diesel Generator System Valve Tables,” Revision 5

Inservice Test Program, “T2300 Primary Containment System Valve Tables,” Revision 5

Inservice Test Program, “T4100 CCHVAC System Valve Tables,” Revision 5

Inservice Test Program, “T4600 Standby Gas Treatment System Valve Tables,” Revision 5

Inservice Test Program, “T4800 Containment Atmosphere System Valve Tables,” Revision 5

Inservice Test Program, “T4901 Pneumatic Supply Division 1 Valve Tables,” Revision 5

Inservice Test Program, “T4901 Pneumatic Supply Division 2 Valve Tables,” Revision 5

Inservice Test Program, “T5000 PC Atmosphere System Valve Tables,” Revision 5

Inservice Test Program, “B3100 Reactor Recirculation System Valve Tables,” Revision 5

Inservice Test Program, “C1100 Control Rod Drive System Valve Tables,” Revision 5

Inservice Test Program, “E1100 & E1150 System Division 2 Valve Tables,” Revision 5

Inservice Test Program, “E1100 & E1150 RHR System Division 1 Valve Tables,” Revision 5

Inservice Test Program, “C5100 Neutron Monitoring System Valve Tables,” Revision 5

Inservice Test Program, “E2100 & E2150 Core Spray System Valve Tables, “ Revision 5

Inservice Test Program, “E4100 HPCI System Valve Tables,” Revision 5

Inservice Test Program, “E5100 RCIC System Valve Tables,”Revision 5

UFSAR 3.3.2.3.4, “Crane and Crane Support Structures,” Revision 8

CARD 03-21688, “Re-Pack P4400F614 in RF10,” dated 10/14/03

WR 000Z031837, “ B3105-F031A N RR Pump Discharge Valve,” dated 4/30/02

WR 000Z000033, “E1150F028B,” dated 11/3/01

WR 000Z012513, “E1150F604A Division 1 MDCT A Inlet ISO Valve,” dated 8/10/01

WR 000Z0012545, “E2150F005B,” dated 5/4/03

WR 000Z014077, “E4150-F003 HPCI Steam Supply OTBD ISO Valve,” dated 11/22/01

WR 000Z012456, “E4150-F600 HCI Steam Supply OTBD ISO BYP Valve,” dated 8/7/01

WR 000Z001744, “E5150-F008 RCIC Steam Line OTBD ISO Valve,” dated 5/5/00

Surveillance procedure 24.707.01, Rev. 39; RWCU valve operability test

Performance evaluation procedure 47.306.01, Rev. 32; Signature analysis of motor-operated valves

Performance evaluation procedure 47.306.02, Rev. 16; VOTES system operating procedure

CECO data for PIS No. G3352F004

CARD 03-11405; G3352F004 does not open after first open signal; dated April 15, 2003

WR 000Z013540; Degraded grease - need to refurb operator; dated October 31, 2001

Selected control room operator logs from December 28, 2003 through December 30, 2003

WR 000Z033636; G3352F220 failed to stroke full open investigate/troubleshoot/repair; dated September 5, 2003

WR 000Z034938; Replace 130VDC HGA relay E11A-K35A; dated December 29, 2003

Maintenance conduct manual MMA11, Rev. 14; Post maintenance testing guidelines

1R22 Surveillance Testing

Procedure 24.202.01, “HPCI Simulated Low RPV Level Automatic Operation and Valve Test,” Revision 74

CARD 03-21611; “A” SBFW pump motor outer bearing TC alarmed and failed to achieve desired flowrate; dated October 21, 2003

DECo File No. S23-348; SBFW pump certified performance curve and test data; dated March 21, 1983

1R23 Temporary Plant Modifications

Temporary Modification 03-0022, “Install Temporary Pump to Take Suction From RB SW Floor Drains Sump G1101D076 and Discharge to Floor Drain G1101D075-21 to Support Maintenance Activities,” dated 10/9/03

Temporary Modification 03-0023, “Install Temporary Pump to Take Suction From RB NW Equipment Drains Sump G1101D073 and Discharge to Floor Drain G1101D075-23 to Support Maintenance Activities,” dated 10/9/03

1EP2 Alert and Notification System (ANS) Testing

Evaluation Report by Analysis and Computing, Inc; FEMA Evaluation of Significant Changes to Alert and Notification Systems

FEMA Approval Letter For Fermi 2 Nuclear Power Station’s Public Alert and Notification System Replacement; dated January 21, 2003

RERP Work Instruction; Maintenance of the Siren Alert Notification System; Revision 1

RERP Work Instruction; Operation of the Siren Alert Notification System; Revision 6

Alert Notification System, Attachment 1; Siren Test Results; July 2002 through September 2003

Maintenance of the Siren Alert Notification System; Requests For Maintenance; May through August 2003

CARD 03-15004; OE 15440 OTH 03015 Inadvertent Activation of a Three Mile Island Alert Notification System Siren Due to Software Problem; dated February 3, 2003

1EP3 Emergency Response Organization (ERO) Augmentation Testing

RERP Plan, Section E; Notification Methods and Procedures; Revision 28

RERP Work Instruction; Emergency Call Out System Maintenance; Revision 10

EP-290; Emergency Notifications; Revision 41

EP-292; Emergency Call Out Backup Method; Revision 24

EP-301-01; Technical Support Center; Revision 16

EP-302-01; Operational Support Center; Revision 12

EP-303-01; Emergency Operations Facility; Revision 10

ECOS Maintenance, Attachment 1; ECOS Test Reviews; May 2001 through September 2003

CARD 03-19620; RERP Drill: Some Members of the ERO Have Privacy Manager on Their Home Phones Which Prevents ECOS From Contacting Them; dated September 9, 2003

CARD 03-19613; RERP Drill: TSC Staff Did Not Meet Recommended 30 Minute Staffing Criteria of EP-301-01; dated September 10, 2003

CARD 01-10184; Key ERO Positions Not Filled During July 24, 2001 ECOS Test; dated July 25, 2001

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

RERP Plan, Section D, HU3 and HA3; Hazards and Other Conditions Affecting Plant Safety; Revision 28

EP-101; Classification of Emergencies; Revision 28

CARD 03-22478; RERP: Potential Violation For Decreasing the Effectiveness of the E-Plan Without Prior NRC Approval; dated November 10, 2003

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies

EP-101, Enclosure A, Tab H, Page H-5 and 6; Hazards and Other Conditions Affecting Plant Safety; Revision 26

Nuclear Quality Assurance Audit Report 03-0107; Emergency Preparedness Program; dated June 13, 2003

Nuclear Quality Assurance Audit Report 02-0105; Emergency Preparedness Program; dated June 17, 2002

Nuclear Quality Assurance Audit Report 02-0104; Self-Assessment of Emergency Response With the Utilities Services Alliance (USA); dated November 14, 2002

Third Tier EP Performance Indicator Data For 2001

RERP Quarterly Program Health Report; dated October 18, 2001

RERP Quarterly Program Health Report; dated August 8, 2001

Memorandum From G. Garber; Drill/Exercise Performance Opportunity Schedule Clarification; dated September 30, 2003

Memorandum From K. Morris; Cycle 03-01 Drill/Exercise Performance Opportunity Schedule; dated January 16, 2003

Memorandum From K. Morris; RERP Program Health Report; dated January 16, 2003

Memorandum From K. Tyger; Self Assessment of Emergency Preparedness With the Utilities Services Alliance (USA); dated November 14, 2002

Quick Hit Self-Assessment; NRC Performance Indicator Drill and Exercise Performance Data Accuracy; dated October 14, 2003

Quick Hit Self-Assessment; Followup to CARD 02-13075; dated February 20, 2003

Preliminary Draft; Emergency Response Critique of the August 14, 2003, Loss of Off-Site Power Unusual Event

Nuclear Plant Event Notification Form; Plant Message Number 1; dated August 14, 2003

Drill/Exercise Critique Summary; August 13, 2003 Radiological Emergency Response Preparedness Drill; dated October 8, 2003

CARD 03-15007; Alternate Emergency Entry and Exit Location Not Identified Procedurally; dated February 27 2003

CARD 03-12375; Consider Adding an Emergency Action Level That Describes a Loss of Torus Under the Loss of Primary Containment Column; dated September 8, 2003

CARD 02-19026; Plant Personnel Evacuation Route Concern; dated August 30, 2002

CARD 02-17180; RERP Exercise: Failure to Meet the Standard For Classification Objective 2.2.1; dated October 2, 2002

CARD 02-16585; HG1 Interpretation Is Not Consistent To Health and Welfare Concerns For the Public; dated July 19, 2002

CARD 02-16305; Loss of Offsite Phone Communications; dated October 19, 2002

CARD 02-13075; RERP: Failure to Meet Business Plan Expectations For Drill and Exercise Performance (DEP); dated April 10, 2002

CARD 02-13073; RERP Objectives Not Met During the March 19th 2002 Drill; dated April 5, 2002

CARD 02-12742; OTH 02-134: OE 14824 - Loss of Emergency Response Organization Telephone Callout System Capability; dated November 7, 2002

CARD 02-11398; RERP: EP101 Procedure Improvement; dated August 20, 2002

CARD 02-11391; RERP: Evaluate RERP Training Program Effectiveness Based On Drill/Exercise Performance; dated September 30, 2002

CARD 02-11385; RERP: Notifications: Inappropriate Actions By TSC Communications During August 27, 2002 Drill; dated September 17, 2002

CARD 02-11355; Audit Finding, Prompt Notification System Siren Replacement Project Actions; dated May 13, 2002

CARD 00-24005; Emergency Preparedness Audit Frequency May Not Be Determined Per 10 CFR 50.54(t) Requirements; dated November 10, 2000

2OS1 Access Control to Radiologically Significant Areas

Radiation Protection CARD Tracking Database (for Cause Codes IP3, IP8, and PR9); dated November 3, 2003

CARD 03-14179; Radiation Protection Gamma Spectroscopy System (Detector 4) Placed in Service After Failing Daily Calibration Check; dated March 11, 2003

CARD 03-14199; Three Workers Externally and Internally Contaminated Working on E1100F050A Valve; dated April 1, 2003

CARD 03-18108; MRP18 and MRP04 Violation, Container of Radioactive Liquid Released from the RRA; dated May 14, 2003

CARD 03-21152; RP In-Use Equipment that Failed Response Check Was Not Tagged Out of Service; dated August 14, 2003

CARD 03-21665; Recommendation for Improvement Regarding RP Postings and Barriers; dated November 4, 2003

MRP06; Accessing and Control of High Radiation, Locked High Radiation, and Very High Radiation Areas; Revision 5

NPRC-03-0281; Self Assessment of Radiation Protection's Human Performance; dated October 28, 2003

PTP 67.000.100; Posting and De-Posting of Radiological Hazards; Revision 11

2OS2 As Low As Is Reasonably Achievable (ALARA) Planning And Controls

2002 Year to Date Departmental Dose (spreadsheet); dated November 3, 2003

2003 Department Dose Status as of 10/1/03; dated October 1, 2003

Active Hot Spot Database; dated November 4, 2003

Fermi 2 Cobalt Reduction Plan (as amended by TMPR memos); dated November 1998

Radiation Protection RF09 Assessment; dated June 29, 2003

CARD 03-14200; Obstacles Encountered Implementing the E11F050A Hydrolazing Work Plan Increased Station Dose; dated April 3, 2003

CARD 03-21664; Audit Finding: Weakness Identified in Radiation Protection's ALARA Practices Based on Review of Several Completed RWP Packages; dated October 24, 2002

FBP-19; Fermi 2 Integrated Work Management Guideline, Appendix G; Revision 4

Post-Job ALARA Review, RWP 03-1103; Drywell - Install and Remove Temporary Shielding, Install Permanent Shielding (EDP 32051); dated June 10, 2003

Post-Job ALARA Review, RWP 03-1105; Install and Remove Scaffold, Power and Lights in Drywell and Steam Tunnel; dated May 22, 2003

Post-Job ALARA Review, RWP 03-1107; Routine Maintenance and I&C Functions and Surveillances including Drywell Equipment Hatch Opening and Closings; dated September 11, 2003

Post-Job ALARA Review, RWP 03-1112; ISI Inspections. Work to Include Welds and Weld Preps, Snubbers and Spring Cans, Bioshield Doors, and All Other Associated Tasks; dated May 22, 2003

Post-Job ALARA Review, RWP 03-1120; SRVs - Remove and Replace, including Interferences and Torquing Flanges as Needed During Vessel Flood Up; dated September 11, 2003

Post-Job ALARA Review, RWP 03-1127; E11FO50A and E11FO50B - Perform Softseat Replacement in Lieu of Hardseat Modifications; dated June 10, 2003

Post-Job ALARA Review, RWP 03-1251; Perform Refuel Activities on RB-5. Includes Vessel Assembly and Disassembly, Core Alterations, ISI Work, Bridge Repair, LPRM Replacement, RP and Radwaste Support of All Activities; dated June 10, 2003

PTP 63.000.200; ALARA Reviews; Revision 14

PTP 63.000.300; Hot Spot Tracking and Removal; Revision 4

TMPR-02-0018; RF09 Scope Review for Cobalt Reduction Program; dated February 27, 2002

TMPR-03-0082; RF10 Scope Review for Cobalt Reduction Program; dated September 22, 2003

WI-RP-002; Work Instruction for the RP Failed Fuel Action Plan; Revision 0

WI-RP-011; Work Instruction for RP Routine Surveys; Revision 1

2OS3 Radiation Monitoring Instrumentation and Protective Equipment

CARD 02-13917; OTH-02-028 Loss of Breathing Air to Bubble Hoods (OE 13365); dated March 6, 2002

PTP 65.000.704; Issuance of Respiratory Protection Equipment; Revision 11

RWP 03-1160; Radiological Pre-Job Briefing Form: Control Rod Drive Exchange; dated April 7, 2003

4OA1 Performance Indicator Verification

Personnel Who Have Received >100 mR for a Single Entry (electronic dosimetry transaction record database for April 2002 - September 2003); dated November 3, 2003

NPRC-02-0221; Radiation Protection NRC Performance Indicators 2$^{nd}$ Quarter 2002; dated July 8, 2002

NPRC-02-0311; Radiation Protection NRC Performance Indicators 3$^{rd}$ Quarter 2002; dated October 11, 2002

NPRC-03-0006; Radiation Protection NRC Performance Indicators 4$^{th}$ Quarter 2002; dated January 10, 2003

NPRC-03-0098; Radiation Protection NRC Performance Indicators 1$^{st}$ Quarter 2003; dated April 10, 2003

NPRC-03-0173; Radiation Protection NRC Performance Indicators 2$^{nd}$ Quarter 2003;

dated July 10, 2003

NPRC-03-0258; Use of Flashing Lights When Accessing Locked High Radiation Area; dated October 3, 2003

NPRC-03-0267; Radiation Protection NRC Performance Indicators 3rd Quarter 2003; dated October 9, 2003

Engineering Support Conduct Manual MES45; Performance Data Reporting; Revision 0

Alert and Notification System Test Records; July 2002 through June 2003

Drill and Exercise Performance Records; July 2002 through June 2003

CARD 03-12428; RERP:  DEP Opportunity Not Counted; dated March 3, 2003

Fermi 2 Team List; dated May 9, 2002

ERO Participation Records; July 2002 through June 2003

4OA2 Identification and Resolution of Problems

Control room logs from October 1, 2003 through December 31, 2003

Procedure 24.321.01, "Dedicated Shutdown Panel H21-P624 Operability Test," Revision 26

Procedure 24.321.02, "Dedicated Shutdown Panel H21-P625 Operability Test, Revision 29

Procedure 24.204.01, "Division 1 LPCI and Suppression Pool Cooling/Spray Pump and Valve Operability Test," Revision 47

Procedure 24.204.05, "Division 1 and 2 RHR Valve Position Indication Verification Test," Revision 33

Procedure 24.204.06, "Division 2 LPCI and Suppression Pool Cooling/Spray Pump and Valve Operability Test," Revision 47

Functional Operating Sketch 6M721-5706-2, "RHR Division 1" Revision W

Work Request 000Z023071, "Investigate, Troubleshoot and Determine Cause of RHR Pump C's CMC Switch Being Stuck," dated 10/9/02

Operator logs first quarter of 2002

Operator logs second quarter of 2002

Operator logs third quarter of 2002

Operator logs fourth quarter of 2002

Performance indicators, “Safety System Unavailability, RHR”

NEI 99-02, “BWR RHR,” Revision 2

Safety tagging records, “Division 1 RHR Torus Spray Valve”

Safety tagging records, “Division 1 RHR Pump C Torus Suction Isolation”

Performance indicators, “Reactor Coolant System Leakage”

NEI 99-02, “Reactor Coolant System Leakage,” Revision 2

Procedure 24.000.02, “Reactor Coolant System Operations Leakage,”

CARD 01-20897; Failure to calculate out of service hours for maintenance rule function; dated December 11, 2001.

CARD 02-11760; Functional failure reviews were not conducted for valve T4500F601; dated April 3, 2002.

CARD 03-18539; Inadequate maintenance rule monitoring; dated July 11, 2003.

CARD 03-23263; System C7100 exceeded its maintenance rule performance criteria; dated December 12, 2003.

CARD 03-12097; EECW flow to E1102C002B(D) seal coolers measured under SOE 03-01 significantly reduced from flows observed in SOE 95-11

GE Nuclear Energy letter No. MHL96052; Residual heat removal pump seal cooling; dated September 18, 1996

E-Mail from Rick Libra to “All engineering managers & all engineering supervisors”; Recent engineering functional analysis trends; dated October 20, 2003

Maintenance work history on E1102C002B & D mechanical seal coolers

CARD 03-11894; RFI #111, NRC SSDI EECW; dated June 5, 2003

Design calculation number DC-5806, Rev. 0; EECW design basis requirements calculation

CARD 00-10052; RHR pump seal water, memo no. MHL96052; dated February 28, 2000

EFA P44-03-007, Rev. A; Div. 1 & 2 EECW hydraulic models were not calibrated following the installation of the new EECW plate and frame heat exchangers

DER 96-1154; Potential problems identified in the defense in depth for RF05; dated September 16, 1996

DER 95-0303; No EECW seal water flow detected to RHR pump D; dated April 27, 1995

CARD 01-13222; Current calibrated Div. 2 EECW hydraulic model may not accurately predict Div. 2 EECW flows; dated July 23, 2001

DECO file No. 01-2541; Letter from Multiple Dynamics Corp to Mr. D. Platt of Borg-Warner; dated September 25, 1984

Design Calculation number DC-4934, Rev. B; Hydraulic analysis, EECW Div. 2; dated December 23, 1993

Design basis document number E11-00, Rev. A; Residual heat removal system

Vendor tech manual number VMR1-45.1, Rev. B; Byron Jackson vertical pump

Vendor tech manual number VMR1-45.2, Rev. F; GE vertical induction triclad motors

DECo file No. R1-7650; GENE 00000022 8680 00; RHR and core spray high temperature pump seal operation report; proprietary document

4OA3 Event Followup

LER 2003-001; Loss of high pressure coolant injection safety function due to closure of steam supply valve; dated August 29, 2003

CARD 03-10957; E4150F003 not stroking closed; dated July 13, 2003

Drawing I-2221-04, Rev. AA; Schematic diagram HPCI sys - steam supply line outboard isolation valves E4150F003, E4150F600

Selected control room operator logs from July 13, 2003 through July 15, 2003

Maintenance rule functional failure evaluation, document ID 1085712; Motor control centers & distribution cabinets; dated July 25, 2003

LER 2003-003; Automatic reactor shutdown due to electric grid disturbance and loss of offsite power; dated October 10, 2003

LER 2003-003, Rev. 1; dated December 17, 2003

CARD 03-19948; Loss of all offsite power due to system grid disturbance; dated August 15, 2003

CARD 03-19464; Inverter failure due to loss of elect grid; dated August 14, 2003

## LIST OF ACRONYMS USED

| | |
|---|---|
| ALARA | As Low As is Reasonably Achievable |
| ANS | Alert and Notification System |
| ANSI/ANS | American National Standard Institute/American Nuclear Society |
| APRM | Average Power Range Monitor |
| CARD | Condition Assessment Resolution Document |
| CCHVAC | Control Center Heating Ventilation and Air Conditioning |
| CFR | Code of Federal Regulations |
| CRC | Curriculum Review Committee |
| CREF | Control Room Emergency Filtration |
| CTG | Combustion Turbine Generator |
| DBA | Design Basis Accident |
| DEP | Drill and Exercise Performance |
| DRP | Division of Reactor Projects |
| DRS | Division of Reactor Safety |
| EAL | Emergency Action Level |
| EDG | Emergency Diesel Generator |
| EECW | Emergency Equipment Cooling Water |
| EESW | Emergency Equipment Service Water |
| EFA | Engineering Functional Analysis |
| EP | Emergency Preparedness |
| E-Plan | Radiological Emergency Response Preparedness Plan |
| ERO | Emergency Response Organization |
| GSW | General Service Water |
| HPCI | High Pressure Coolant Injection |
| HRA | High Radiation Area |
| IP | Inspection Procedure |
| JPM | Job Performance Measure |
| LER | Licensee Event Report |
| LMTD | Logarithmic Mean Temperature Difference |
| LOCA | Loss of Coolant Accident |
| LOOP | Loss of Offsite Power |
| LORT | Licensed Operator Requalification Training |
| MG | Motor Generator |
| M&TE | Measuring and Test Equipment |
| NCV | Non-Cited Violation |
| NEI | Nuclear Energy Institute |
| NRC | Nuclear Regulatory Commission |
| NRR | Office of Nuclear Reactor Regulation |
| ODCM | Off-site Dose Calculation Manual |
| PI | Performance Indicator |
| PMT | Post Maintenance Testing |
| RCIC | Reactor Core Isolation Cooling |
| RERP | Radiological Emergency Response Preparedness |
| RHR | Residual Heat Removal |
| RHRSW | Residual Heat Removal Service Water |
| RO | Reactor Operator |

| | |
|---|---|
| RP | Radiation Protection |
| RWP | Radiation Work Permit |
| SDP | Significance Determination Process |
| SRO | Senior Reactor Operator |
| UFSAR | Updated Final Safety Analysis Report |
| UHS | Ultimate Heat Sink |
| VHRA | Very High Radiation Area |