October 28, 2003

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 05000341/2003008

Dear Mr. O'Connor:

On September 30, 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 October 3, 2003, with you, Mr. Cobb, and other members of your staff.

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

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

If you contest the subject or severity of 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 made 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
Branch 1
Division of Reactor Projects

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

Enclosure: Inspection Report 05000341/2003008
w/Attachment: Supplemental Information

cc w/encl: N. Peterson, Director, Nuclear Licensing
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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-341
License No:	DPR-43
Report No:	05000341/2003008
Licensee:	Detroit Edison Company
Facility:	Enrico Fermi, Unit 2
Location:	6400 N. Dixie Hwy. Newport, MI 48166
Dates:	July 1 through September 30, 2003
Inspectors:	S. Campbell, Senior Resident Inspector T. Steadham, Resident Inspector T. Ploski, Senior Emergency Preparedness Inspector R. Walton, Reactor Inspector R. D. Alexander, Radiation Specialist M. Mitchell, Reactor Engineer
Approved by:	Mark Ring, Chief Branch 1 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000341/2003008; 07/01/2003 - 09/30/2003; Fermi Nuclear Power Station, Unit 2; Event Follow-up, and Other Activities.

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

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Non-Cited Violation of Criterion III of Appendix B to 10 CFR Part 50, for failure to assure adequate design controls were in place to ensure that Agastat general purpose relays would be replaced prior to exceeding their design basis life. Although the licensee's preventive maintenance program allowed safety-related general purpose relays to remain in service beyond their design basis life, a review of work history identified no general purpose relays that had malfunctioned due to heat-related problems.

  This finding is greater than minor because, if left uncorrected, it would become a more significant safety concern. Specifically, the licensee's process of including a 25 percent grace period on most preventive maintenance tasks could allow a component to remain in service longer than the design basis lifetime, thus reducing the reliability of that component to perform its intended safety function. Because the relay that was found in service beyond its design basis lifetime remained functional, this finding did not represent an actual loss of a safety function. Therefore, this finding is characterized as an issue of very low safety significance. (Section 4OA3)

- Green. The inspectors identified a Non-Cited Violation of Criterion III of Appendix B to 10 CFR Part 50, for site personnel installing plastic sleeves on the drain lines for all four emergency diesel generators without using the design control measures for design changes specified in Procedure MES 12, "Temporary Modifications." Consequently, installation of the plastic sleeves for the drain line on Emergency Diesel Generator 11 restricted the oil draining capacity of the diesel and was a contributing cause for oil reaching the hot exhaust manifold and creating a fire.

  This finding is greater than minor because it affected the Mitigating System Cornerstone of equipment reliability. Specifically, the plastic sleeves restricted the fuel oil draining flow for Emergency Diesel Generator 11. The restriction caused the fuel oil to collect on the injector deck, migrate, and collect on the hot exhaust manifold piping insulation and catch fire. The finding is of very low safety significance because the fire was manually suppressed using available fire extinguishers before substantial damage to Emergency Diesel Generator 11 occurred. Also, emergency onsite power availability was maintained in that only one of four emergency diesel generators was impacted. (Section 4OA3)

B. Licensee-Identified Violations

No findings of significance were identified.

REPORT DETAILS

Summary of Plant Status

Fermi 2 began this inspection period at 100 percent rated thermal power until 4:10 p.m. on August 14, 2003, when a loss of offsite power caused a turbine trip and a reactor SCRAM. The reactor remained shutdown in Mode 3 to complete system lineups and testing. At 5:25 p.m. on August 18, 2003, the operations staff commenced withdrawing control rods for reactor startup. The reactor was declared critical at 5:07 p.m. later that day. The reactor reached 100 percent rated thermal power at 6:30 p.m. on August 21, 2003, but power was reduced to 83 percent for rod pattern adjustment the next day.

Power was again raised to 100 percent a few hours later and it remained at or near that level until September 5, 2003, when the reactor was shutdown to facilitate repairs on a leaking reactor water cleanup valve (G33F120). At 1:13 a.m. on September 9, 2003, the operations staff commenced withdrawing control rods for reactor startup. The reactor was declared critical a few hours later at 4:41 a.m. The reactor reached 100 percent rated thermal power at 4:53 p.m. on September 11, 2003, and it remained at or near that level for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstone: Mitigating Systems and Barrier Integrity

1R01 Adverse Weather (71111.01)

a. Inspection Scope

The inspectors selected three risk-significant systems (the residual heat removal [RHR] complex, condensate return and storage tanks, and the general service water complex) that are required to be protected from adverse hot 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 hot weather conditions.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments

.1 Partial Walkdowns (71111.04Q)

a. Inspection Scope

The inspectors performed 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 Condensate (P1100) system.

b. Findings

No findings of significance were identified.

.2 Complete Walkdown (71111.04S)

a. Inspection Scope

The inspectors performed a complete walkdown of the standby liquid control (C4100) system. This system was selected because of its risk-significance in the licensee's probabilistic risk assessment. The inspection reviewed the following:

- appropriate plant procedures;
- standby liquid control system drawings;
- UFSAR to identify proper system alignment;
- system training manual;
- maintenance work requests;
- configuration control condition reports;
- outstanding design issues and operator work arounds;
- control room logs;
- design basis documents;
- vendor technical manuals;
- system health report; and
- chemistry sample results.

Combined with an electrical and mechanical walkdown, the inspectors used the documents to verify valves were positioned correctly and did not exhibit leakage that would impact the valve's function, availability of electrical power, proper labeling, lubrication and cooling of major equipment, functionality of hangers and support systems, equipment was being properly maintained, and that appropriate surveillances and other required tests were being adequately performed.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

a. Inspection Scope

The inspectors toured the following areas to determine whether combustible hazards were present, fire extinguishers were properly filled and tested, the CARDOX units were operable, hose stations were properly maintained, and if the fire hazard analysis drawings were correct:

- Division 2 switchgear room, UFSAR 9A.4.2.5, Zone 4;
- Division 1 switchgear room, UFSAR 9A.4.2.13, Zone 12;
- Division 1 battery room, UFSAR 9A.4.2.11, Zone 10;
- Division 2 battery room, UFSAR 9A.4.2.11, Zone 10;
- Division 1 control center heating ventilation air conditioning UFSAR 9A.4.2.15, Zone 14;
- Division 2 control center heating ventilation air conditioning, UFSAR 9A.4.2.15, Zone 14;
- Relay room, UFSAR 9A4.2.4;
- General service water complex, UFSAR 9A4.8;
- Motor generator set room UFSAR 9A4.1.9;
- Emergency equipment circulating water (EECW), UFSAR 9A4.1.7;
- 1st Floor turbine building;
- 2 nd Floor turbine building;
- 3rd Floor turbine building;
- Division 1RHR service water pump room, UFSAR 9A.4.3; and
- Division 2 RHR service water pump room, UFSAR 9A.4.3.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

The inspectors observed licensee training personnel evaluate two operating crews during an accident scenario and subsequently observed the training personnel critique the operating crews' performance. The inspectors observed crew for communications, alarm response, emergency operating procedure usage, component operations and emergency plan classifications. The inspectors reviewed the scenario for operational validity and appropriate selection of critical tasks. The inspectors discussed scenario observations and crew evaluations with the licensee trainers.

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 and the control room unit logs for the following systems:

- primary containment isolation valves (A7100);
- high pressure coolant injection system (E4100); and
- RHR (E1100).

The inspectors independently evaluated the licensee's determination of maintenance rule functional failures, and reviewed surveillance procedures and operators' logs to assess the licensee calculation of system unavailability. The inspectors also evaluated the results of the licensee's Appendix J (Primary Reactor Containment Leakage Tests) associated with the A7100 system. 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 selected documents, listed in the "List of Documents Reviewed" section of this report, to determine if the risk associated with the activities listed below agreed with the results provided by the licensee's risk assessment tool. In each case, 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. The inspectors assessed the following maintenance activities:

- broken high pressure coolant injection drain pot;
- maintenance risk for RHR outage;
- emergent work on combustion gas turbine generator (CTG) 11-3 during CTG 11-1 inspection;
- forced outage 03-01, "Loss of Offsite Power;" and
- water leak on reactor water cleanup check valve G33F120.

b. Findings

No findings of significance were identified.

1R14 Nonroutine Plant Evolutions (71111.14)

.1 Station Blackout Combustion Turbine Generator (CTG) Failure to Start During Loss of Electrical Grid

a. Inspection Scope

The inspectors reviewed CARD 03-19464, Work Requests (WR) 000Z033553 and 000Z033258 and Technical Service Request 32666 and conducted interviews with operations and system engineering personnel to determine the circumstances surrounding the failure of CTG 11-1 to start during the loss of offsite power (LOOP) that occurred on August 14, 2003. The inspectors reviewed this event to determine whether failure of CTG 11-1 to start was attributed to human error.

b. Findings

Seven to eight minutes after the plant experienced a LOOP on August 14, 2003, the operators attempted to start the Station Blackout CTG 11-1. The starting diesel was running and the CTG turbine shaft was spinning but the CTG was not firing. Combustion Turbine Generator 11-1 was not firing because the inverter, which is used for powering the two igniters on the CTG, had lost power. Several attempts to restart the inverter were unsuccessful. A shift manager authorized using a portable 120VAC generator to supply power for starting the inverter. An operator connected the portable generator and powered the inverter. Two additional attempts were made for starting CTG 11-1 but the CTG tripped at partial speed due to a “loss of flame” and a “failure to ignite” signal. These signals were generated because the contactor for an emergency fuel forwarding pump failed. While holding the contactor together by hand, the emergency fuel forwarding pump was started and a third attempt at starting CTG 11-1 was successful. The licensee initiated CARD 03-19646 to document the condition.

Troubleshooting of the emergency fuel forwarding pump was performed under WR 000Z033553. Electricians found that the contactor was hanging up on the lower portion of an arcing horn (used for arc suppression drain) which prevented the contact from fully closing. An adjustment to the arcing horn clearance was made to eliminate the interference and the emergency fuel forwarding pump worked.

Troubleshooting of the inverter was performed on August 15, 2003, under WR 000Z033258. The licensee discovered that the low voltage trip setpoint of the inverter was set at 105 VDC. This value was based on CTG 11-1 having a 60 cell battery (1.75 VDC per cell). Actually, CTG 11-1 has 56 cells. Visicorder traces for a blackstart test of CTG 11-1 conducted under Sequence of Events 96-12 on November 9 and 10, 1996, had indicated a momentary voltage dip of 105 VDC during CTG 11-1 start. A similar test was performed on September 3, 2003, and low voltage dip was measured below the 105.23 VDC setpoint at 104.6 VDC. Engineering personnel initiated Technical Service Request 32666 to adjust the low voltage trip setpoint to between 93.1 and 93.6 VDC. The technical service request was implemented on September 4, 2003, under WR 000Z033258.

Combustion Turbine Generator 11-1 was installed under the requirements of 10 CFR 50.63, “Loss of all Alternating Current,” or a station blackout event. The CTG did not function when called upon during the loss of offsite power event that occurred on August 14, 2003, and would not have fulfilled its safety function of supplying alternating current during a station blackout event. Additionally, the licensee determined this issue to be a Maintenance Rule Functional Failure as documented in Evaluation ID 030814-2. This issue is an Unresolved Resolved Item **(URI 0500341/2003008-01)** pending the inspectors’ review of the basis for the inverter low voltage trip setpoint set above the actual voltage dip during CTG 11-1 starting and why previous operability tests of CTG 11-1 failed to identify this condition before the August 14, 2003, event.

.2 Repair of Reactor Water Cleanup Check Valve, G33F120

a. Inspection Scope

During the plant restart following the August 14, 2003, LOOP, a leak developed on a reactor water cleanup check valve. Hot torquing the valve, which was expected to seal the leak did not work. The licensee later decided to shut the unit down to repair the valve. The inspectors reviewed WRs, various plant procedures, applicable technical specifications, drawings, and interviewed personnel involved with this event. The inspectors reviewed the evolution and operator response to assess human performance issues.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors assessed several evaluations or issues associated with equipment operability. The following samples were reviewed to ensure that operability was properly justified and the component or system remained available, such that no unrecognized increase in risk occurred:

- Degraded fire seal in Control Room, CARD 03-10893;
- Potential inadequate linear heat generation rate in single loop operation, Engineering Functional Analysis--J11-01-011;
- Removal of reactor core isolation cooling floor plug, Engineering Functional Analysis-E51-03-013;
- Past operability on low EECW flow, CARD 02-16602;
- G3300F120, "Mechanical Joint Evaluation," Log 03-0037; and
- EECW starting transient on 9/23/03, CARD 03-21938 and Engineering Functional Analysis P44-03-015.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

Engineering Design Package 30251, for upgrading the integrated plant computer system, was reviewed and selected aspects were discussed with engineering personnel. This document and related documentation were reviewed for adequacy of the safety evaluation and consideration of design parameters. The modification was for equipment upgrades of existing equipment.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed and observed post maintenance testing of the reactor building heating, ventilating, and air conditioning vent system particulate iodine noble gas (SPING) activities involving risk significant equipment in the Mitigating Systems cornerstone.

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.

1R20 Refueling and Outage (71111.20)

a. Inspection Scope

The inspectors observed control room activities associated with the shutdowns resulting from the August 14, 2003, LOOP and the subsequent forced outage to repair a leaking check valve. The inspectors reviewed various activities including inserting and withdrawing control rods, monitoring cool-down and heat-up rates, restoration of offsite electrical lines, spent fuel cooling system operation, and completing mode specific surveillance testing. The inspectors also verified defense-in-depth for shutdown cooling, performed periodic panel walkdowns in the control room, and attended licensee outage planning meetings.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed surveillance testing activities and/or reviewed completed packages for the tests listed below related to systems in the Mitigating Systems and Barrier Integrity Cornerstones:

- Procedure 44.020.025 - Main Steam Line Radiation Monitor A Calibration/Functional;
- Procedure 44.010.108 - Intermediate Range Monitor Calibration;
- Control Rod Friction Test; and
- Procedure 24.307.36, Diesel Generator Service Water 13 Flow Operability Test (Reviewed Package).

The inspectors verified that the structures, systems, and components selected were capable of performing their intended safety function and that the surveillance tests satisfied the requirements contained in Technical Specifications, the UFSAR, and licensee procedures. During surveillance testing observations, the inspectors verified that the test demonstrated operational readiness consistent with design and licensing basis documents and that the test acceptance criteria were clear. The inspectors also verified that the impact of the testing had been properly characterized during the pre-job briefing; the test was performed as written; the test data was complete and met the requirements of the testing procedure; and the test equipment range and accuracy was consistent with the application. Following test completion, the inspectors verified that the test equipment was removed and that the system was returned to its normal standby configuration.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed documentation for a temporary configuration change for Temporary Modification 03-0020, “Jumper Safety Relief Valve A Tailpipe Pressure Switch.”

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

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

a. Inspection Scope

The inspectors reviewed Revisions 27 and 28 of the Fermi Nuclear Power Plant Emergency Plan to determine whether changes identified in Revisions 27 and 28 reduced the effectiveness of the licensee's emergency planning, pending on-site inspection of the implementation of these changes.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed the Blue Team emergency preparedness drill conducted on July 30, 2003, for weaknesses and deficiencies in classification, notification and protective action requirement development activities. The inspectors compared observed identified weaknesses and deficiencies against licensee identified findings to determine whether the licensee was properly identifying drill performance issues. The inspectors determined whether the licensee's assessment of performance was per the applicable criteria. The inspectors used guidance document NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," during the inspection.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety (PS)

2PS2 Radioactive Material Processing and Transportation (71122.02)

.1 Radioactive Waste System Description and Waste Generation

a. Inspection Scope

The inspectors reviewed the liquid and solid radioactive waste system description in the Updated/Final Safety Analysis Report (UFSAR) and the 2002 Radiological Environmental Operating Report for information on the types and amounts of radioactive waste (radwaste) generated and disposed.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Radioactive Waste System Walkdowns

a. Inspection Scope

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

Though there were no changes to the waste processing system since the last inspection in this area, the inspectors discussed with licensee staff plans for changes to the system and the depth of the evaluations (i.e., 10 CFR 50.59 evaluations) anticipated to be completed in conjunction with the modifications. The inspectors reviewed the current processes for transferring waste resin into shipping containers to determine if appropriate waste stream mixing and/or sampling procedures were utilized. The inspectors also reviewed the methodologies for waste concentration averaging to determine if representative samples of the waste product were provided for the purposes of waste classification in 10 CFR 61.55.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.3 Waste Characterization and Classification

a. Inspection Scope

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

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.4 Shipment Preparation and Shipping Records

a. Inspection Scope

The inspectors reviewed the documentation for shipment packaging, surveying, labeling, marking, placarding, vehicle checks, emergency instructions, disposal manifest, shipping papers provided to the driver, and licensee verification of shipment readiness for eight, non-excepted radioactive material and waste shipments during calendar years 2002 and 2003. These shipments included:

- Three High Integrity Containers of Dewatered Resin to Barnwell, SC (all Low Specific Activity (LSA)-II);
- Contaminated Laundry to Unitech in Morris, IL (LSA-II);
- Steam Relief Valves (SRV) & SRV Pilots to NWS Technologies in Spartenburg, SC (Surface Contaminated Object (SCO)-II);
- High Radiation DAW to Duratek in Oak Ridge, TN (LSA-II);
- CD-600 Dryer & RF-22 Refrigerant to ALARON Corp in Wampum, PA (SCO-II and Non-Flammable Compressed Gas (Class 2.2)); and
- 10 CFR Part 61 (Waste Stream) Samples to Duke Engineering in Westborough, MA (Type A).

The inspectors verified that the requirements of any applicable transport cask Certificate of Compliance were met and verified that the receiving licensee was authorized to receive the shipment packages. The inspectors verified that the licensee's procedures for cask loading and closure procedures were consistent with the vendor's approved procedures. As there were no licensee shipment activities conducted during the inspection, the inspectors reviewed the training material (i.e., lesson plans) provided to personnel responsible for the conduct of radioactive waste processing and radioactive shipment preparation activities. The review was conducted to verify that the licensee's training program provided training consistent with NRC and Department of Transportation requirements.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

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

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

1. Initial problem identification, characterization, and tracking.
2. Disposition of operability/reportability issues.
3. Evaluation of safety significance/risk and priority for resolution.
4. Identification of repetitive problems.
5. Identification of contributing causes.
6. Identification and implementation of effective corrective actions.
7. Resolution of non-cited violations (NCVs) tracked in corrective action system(s).
8. Implementation/consideration of risk significant operational experience feedback.

Finally, the inspectors reviewed the scope of the licensee's audit program with regard to radioactive material processing and transportation programs to verify that it met the requirements of 10 CFR 20.1101(c).

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

Cornerstones: Barrier Integrity and Public Radiation Safety

.1 Reactor Safety Strategic Area

a. Inspection Scope

The inspectors sampled the licensee's submittals for performance indicators (PIs) and periods 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:

- Reactor Coolant System Specific Activity

  Specifically, the inspectors reviewed the licensee's assessment of this PI by reviewing Chemistry Department records and selected isotopic analyses (January 2002 through June 2003) to verify that the greatest Dose Equivalent Iodine (DEI) value obtained during those months corresponded with the value reported to the NRC. The inspectors also reviewed selected DEI calculations to verify that the appropriate conversion factors were used in the assessment as required by Technical Specifications. Additionally, on September 10, 2003, the inspectors observed a chemistry technician obtain and analyze a reactor coolant sample for DEI to verify adherence with licensee procedures for the collection and analysis of reactor coolant system samples.

b. Findings

No findings of significance were identified.

.2 Radiation Safety Strategic Area

a. Inspection Scope

The inspectors sampled the licensee's submittals for PIs and periods 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:

- RETS/ODCM Radiological Effluent Occurrences

  Specifically, the inspectors reviewed the licensee's dose records related to both liquid and gaseous effluent releases from the station from October 2002 to June 2003, to determine if this data was adequately assessed and reported. Since no reportable events were identified by the licensee for these time periods, the inspectors compared the licensee's data with the corrective action program database 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.

4OA2 Identification and Resolution of Problems (71152)

a. Inspection Scope

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 level, that corrective actions were performed in a timely manner and that adverse trends were identified and addressed. The inspectors selected the following issues to determine if problem characterization was accurate and to verify that extent of condition reviews were adequately completed or were in the process of being performed: opening and closing problems of the reactor core isolation cooling (RCIC) turbine governor control valve, E5150F044, since 1999.

b. Findings

No findings of significance were identified.

4OA3 Event Followup (71153)

.1 Failure to Apply Adequate Design Control Measures Results in Equipment Exceeding Its Design Basis Service Life.

a. Inspection Scope

The inspectors reviewed plant maintenance history, engineering calculations and analyses, completed work orders, industry experience, other licensee documentation, and interviewed personnel to determine if the licensee's preventative maintenance (PM) program was effective in ensuring that equipment is replaced prior to the end of its design basis service life.

b. Findings

Introduction: The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion III having very low safety significance (Green) for failing to apply adequate design control measures.

Description: On June 27, 2003, a technician was replacing the reactor vessel high steam line pressure relay and inadvertently removed the wrong relay. While reviewing this event, the inspectors determined that although the relay had a design basis service life of 4.5 years, the relay had been in service for 5.6 years.

Information Notice 84-20 informed the licensee that based on General Electric test data, the service life of all Agastat general purpose relays was 4.5 years. In memorandum NE-NS-87-0075, the licensee adopted the 4.5 year service life and identified the reactor vessel high steam line pressure relay as one of approximately 180 affected relays. Although this memo stated that the relays must be replaced "before their 4.5 year life span has expired," the newly-created preventive maintenance tasks included a 25 percent grace period on many of the relays identified in the memo.

The licensee currently has 238 normally-energized Agastat general purpose relays in safety-related applications. Historically, many of these relays have been replaced within their grace period. The use of these relays beyond their service life was caused by an expectation that allowed routine maintenance tasks to be completed between the due date and the critical date. The inspectors concluded that the grace period was inappropriate because it allowed these relays to exceed their analyzed service life without an adequate engineering justification.

Analysis: The inspectors determined that failure to ensure that safety-related Agastat series GP relays were replaced prior to exceeding their design basis service life was a performance deficiency warranting a significance evaluation. 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 failure to apply proper design control measures, if left uncorrected, would become a more significant safety concern. Specifically, automatically including a 25 percent grace period on most preventive maintenance (PM) tasks could allow other safety-related components to exceed a similar service schedule based on an engineering analysis.

In response to this issue, the licensee sent the recently-removed reactor vessel high steam line pressure relay to an outside vendor for testing. This test showed that the relay was functional when it was removed. Additionally, the licensee performed an Arrhenius calculation for this relay based upon relevant environmental conditions. The calculation concluded that the relay had an expected service life of 8.95 years.

A review of all 238 normally-energized Agastat general purpose relays in safety related applications identified five that were currently within their grace period. Since the sole function of all five relays was for alarm annunciation in the control room, their failure would not represent an actual loss of a safety function. Furthermore, these relays were subjected to essentially the same environmental conditions as the reactor vessel high steam line pressure relay and would therefore be expected to age in a similar manner. The inspectors concluded that there was reasonable assurance that these five relays would not fail prior to the expiration of their grace period.

Therefore, this finding does not represent an actual loss of a safety function. 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. The inspectors concluded that the issue was of very low safety significance.

Enforcement: 10 CFR 50, Appendix B, Criterion III, states, in part, that design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design. Contrary to the above, the licensee failed to apply design control measures commensurate with the previous Agastat Series GP Relay service life calculations including documentation of the licensee's basis for changing the service life from 4.5 to 5.625 years.

The licensee has since removed any grace period on these relays, is currently re-analyzing their service life, and is reviewing the PM program to determine proper application of the 25 percent grace period to other equipment. Because this failure to apply design control measures is of very low safety significance and has been entered into the licensee's corrective action program as CARDs 03-10985 and 03-19313, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000341/2003008-02).

.2 August 14, 2003 Loss of Offsite Power

a. Inspection Scope

On August 14, 2003, the plant experienced a complete loss of offsite power with subsequent turbine trip and reactor SCRAM. From the control room, the inspectors monitored plant conditions and operator actions to ensure that the plant was responding as designed, that all relevant procedures were being followed. The inspectors walked down the control panels, reviewed various procedures, drawings, Technical Specifications, and other licensee documentation, and interviewed various plant personnel. The inspectors reviewed the Transient Analysis Program report and compared plant response to the expected response as detailed in the UFSAR.

b. Findings

No findings of significance were identified.

.3 (Closed) URI 50-341/03-002-01 "Fire on Emergency Diesel Generator (EDG) 11 Exhaust Pipe:"

Introduction: One Green NCV was self-revealed for failure to comply with 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) for installing temporary drain lines on all four EDGs on January 17, 2003, without following the temporary modification process.

Description: During the EDG 11 post maintenance test run per procedure 23.307 on January 31, 2003, the licensee noticed fuel oil spilling from the clean fuel drain header vent (J-tube) onto the injector deck. The oil migrated from the deck onto the hot exhaust manifold. The high exhaust manifold temperature ignited the fuel oil on the control side (side with governor) and opposite control side (side without governor) of EDG 11. Plant personnel extinguished the fire and operators shut down the diesel.

Condition Assessment Resolution Document 03-00518 was written to document the event. A root cause team was formed and the team determined that the root cause of the fuel oil spilling from the J-tube was that the clean fuel drain flow line capacity was restricted due to three contributing causes:

- Improper torque applied to the adjusting plug during testing and/or reassembly of the control side cylinder No. 2, injector nozzle 34408 resulted in excessive fuel oil injector leak-off to the clean fuel drain header;
- Partial internal blockage (debris) of the clean fuel drain line; and
- Partial restriction at the end of the clean fuel drain line due to installation of temporary plastic sleeves.

Mechanics replaced fuel injectors during performance of the 18-month PM on January 30, 2003, under work package W836030100 per procedure 34.307.001, "Emergency Diesel Generators - Inspection and Preventive Maintenance." After the event and during a February 6, 2003, run of the diesel, engineers noticed that EDG 11 cylinder number 2 exhaust temperatures were about 100 degrees Fahrenheit below normal. This decrease in temperature was attributed to injector nozzle 34408 allowing more fuel oil flow through the nozzle.

Nozzle 34408 was sent to the vendor for failure analysis where a "pop" test of the nozzle revealed an improper torque on the spring sleeve which allowed too much fuel oil to pass through the injector. Although the torque should have been 55 ft-lbs, the as-found torque was 20 ft-lbs. The licensee later determined that procedure 34.307.001 provided insufficient information to properly set the injector nozzle spring sleeve torque.

Partial internal blockage was another possibility identified by the root cause team; however, since the drain line was blown down with high pressure air shortly after the event, this possibility could not be substantiated.

The last cause identified by the root cause team was the installation of temporary sleeves on the end of the fuel oil drain line. A cracked, buried drain line running from the RHR complex to the chemical pond created a direct path for draining diesel fuel oil to ground water. To prevent this condition, plastic sleeves were installed on the drains of all four EDGs and routed to aluminum pans to collect oil drained during a diesel run. The root cause team determined that the plastic sleeves were not installed in accordance with procedure MES-12, "Temporary Modifications." Consequently, the licensee did not adequately assess the installation of the temporary sleeves and the effect the sleeves would have on restricting flow of fuel oil through the drain line.

Initially, station personnel thought that the temporary sleeves restricted oil draining, backed up in the header through the vent and migrated to the exhaust manifold causing the fire. Event Notice 39554, communicated to the NRC on February 3, 2003, documented that all four EDGs were declared inoperable following discovery of a fuel oil leakage drain path change that was installed to prevent fuel oil leakage from reaching the ground water. In response, the licensee removed the plastic sleeves from the drains and declared the EDGs operable. However, as mentioned above, the root cause team later determined that the sleeves were one of three contributing causes.

Analysis: The Inspectors considered these three issues individually. Although the first issue involved an inadequate procedure, it represented just one of several ways that an excess amount of fuel oil could have been drained. Had the drain line not been restricted, then the line would have been of sufficient size to discharge the increased fuel oil flow without causing the oil to back up in the line. Since improperly setting the spring sleeve torque would not, in and of itself, have caused the fire, this issue was considered minor.

Second, since debris in the line was an unproven theory, this could not be considered a finding. Shortly after the event, the licensee blew down the clean oil drain line to remove any debris that might have accumulated in the line. Since any debris cleared was not collected there is no way to determine if the line was blocked and, if it was, how it became blocked. Therefore, the inspectors could not identify any performance deficiency associated with this possible root cause.

Third, the inspectors determined that the failure to adequately assess the installation of the temporary sleeves and the effect it would have on restricting flow of fuel oil through the drain line was a performance deficiency warranting a significance evaluation. 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 if left uncorrected, the finding would become a more significant safety concern. Specifically, by restricting the flow, any event that increased the oil flow through the drain line could have caused a fire.

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 fire caused a failure of only EDG 11 and all other EDGs remained available, the safety function of providing emergency electrical power was not lost. The inspectors therefore concluded that the issue was of very low safety significance.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, states, in part, that design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design and be approved by the organization that performed the original design unless the applicant designates another responsible organization. Contrary to the above, site personnel installed plastic sleeves on the drain lines for all four EDGs without using the design control measures for design changes specified in Procedure MES 12, "Temporary Modifications." Consequently, the installation of plastic sleeves on the drain line of EDG 11 restricted the oil draining capacity of the diesel and was a contributing cause for oil reaching the hot exhaust manifold and creating a fire. Because the failure to implement design control processes for adding the plastic sleeves is of very low safety significance and has been entered into the licensee's corrective action program (CARD 03-00518), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000341/2003008-03).

.4 (Closed) URI 50-341/2002-011-001, "Low Flow Conditions On the Division 2 Emergency Equipment Cooling Water (EECW) System:"

This item involved repetitive low flow conditions that the licensee discovered on the Division 2 EECW system. The licensee wrote several CARDs addressing this issue. The licensee flushed the RHR cooler, the thermal recombiner cooler and the battery room cooler and found black iron oxide corrosion. Black iron oxide had been found in a leaking non-interruptible air supply room cooler during Refueling Outage 7 while repairing the leaking cooler. The licensee stated that no instances of low flow conditions occurred as a result of corrosion in the EECW system in the past.

The inspectors reviewed corrective action documents written since January 1, 1990, to determine whether previous occurrences of corrosion induced low flow conditions had occurred on the EECW system. Other than the CARDs written documenting the most recent issues, no occurrences of corrosion induced low flow conditions were identified during that time period.

The inspectors reviewed the resolution to the corrosion issue. Condition Assessment Resolution Document 02-19698 was written to evaluate chemical treatment of the EECW system. Chemists planned to add General Electric Betz Corsheild MD4103 chemical, which pacifies the metal oxide layer to protect the pipes/tubes and inhibits copper corrosion, by the end of October 2003. Inhibitor should be added once. The licensee considered other inhibitors but, due to environmental concerns of draining the drywell coolers to the floor drain and the manpower needed to direct the water to barrels, the licensee decided to install in-line demineralizer cartridges to remove the inhibitor from the water, thereby alleviating environmental concerns.

The inspectors reviewed the past operability for low flow conditions documented in CARD 02-16620. The past operability review used design calculations that documented degraded flow conditions for equipment subjected to loss of coolant accident (LOCA) and high energy line break heat loads and for evaluating environmental qualification of safety-related electrical components. In some instances, the "WTRCOIL" program was used to determine whether the as-found low flow conditions were below design limits. The past operability summarized the following:

- For RCIC and the Division 1 core spray room cooler T4100B021, the worst case flow condition (non-essential loads open and degraded pump conditions) in Design Calculation (DC) 5806 was 80.81 gpm, which bounded the as-found flow of 83 gpm documented in CARD 03-12942.

- For the recombiner space cooler T4100B036, the as-found flow of 19.5 gpm was slightly bounded by the design degraded flow condition of 19.4 gpm (non-essential loads open and degraded flows) documented in DC 5806.

- For the switchgear space cooler T4100B003, the as-found flow documented in CARD 03-12942 was 26.4 gpm which was below the design minimum with the non-essential loads restored. The minimum design flow was 27 gpm. For this component, the "WTRCOIL" program was run and the licensee determined that 26.4 gpm exactly matched the design flow minimum (no margin for error).

- Per DC-5806, the design flow requirement for the battery charger space cooler T4100B003 was 20 gpm, which bounded the as found flow of 23.3 gpm documented in CARD 02-19278.

- Per DC-5589, the design flow requirements for the RHR space room cooler T4100B019 was 136.8 gpm with non-essential loads isolated (high energy line break) and 205.6 gpm (LOCA). However, per DC-5888, this calculation documented that restoration of the non-essential loads reduces the required deliverable (degraded) flow from 136.8 gpm to 121.5 gpm which bounded the as-found flow of 129.2 gpm documented in CARD 03-12900.

- For the core spray space room cooler T4100B020, the Sargent and Lundy analysis predicted post LOCA temperatures in this room to be less than the original General Electric qualification temperature of 148 degrees Fahrenheit.

- Sargent and Lundy analysis predicted peak post LOCA temperatures in the recombiner space room cooled by cooler T4100B037 to be less than the original General Electric qualification temperature of 148 degrees Fahrenheit.

- The as-found flow value for the switchgear space room cooler T4100B004 was within range and no evaluation was needed.

- The program “WTRCOIL” was run for the battery charger space room cooler T4100B044, which had an as-found flow documented on CARD 02-16202 of 1.5 gpm. At this flow rate, the space served by this cooler would have remained below 127 degrees Fahrenheit assuming EECW was at a temperature of 95 degrees Fahrenheit. This would have exceeded the 122 degrees Fahrenheit maximum design temperature for the battery charger. Per DC 5406, the battery charger equipment would operate at a maximum temperature of 140 degrees Fahrenheit for 24 hours without operator action.

- The as-found flow value for the switchgear space room cooler T4100B005 was within range and no evaluation was needed.

Although in some cases, the as-found flows were determined to be very marginal, the licensee had adequately demonstrated that systems cooled by the EECW system would have performed their intended safety function under degraded EECW flow conditions. This item is closed.

.5 <u>(Closed) 50-341/2003-002-02, “Testing of Buried Ethylene Propylene Rubber (EPR) Cables</u>

This item involved safety-related EPR cables buried underground whose insulation would be subjected to moisture intrusion (i.e. submergence). Several industry issues involving EPR cable faults due to moisture intrusion have been documented. NUREG-1801, Generic Aging Lessons Learned Report (NUREG-1801, Volumes 1 and 2) stated that when an energized medium-voltage cable is exposed to wet conditions for which it is not designed, water treeing or a decrease in the electric strength of the conductor or insulation can occur and potentially lead to failure. Given the existence of NUREG 1801 and the industry issues, the inspectors expected that the licensee would have established testing programs to verify the integrity of the cables.

One opportunity existed to address this issue previously. The inspectors found CARD 99-18796 written on December 17, 1999, to assess applicability of the failure of underground 5 kV cables at Davis-Besse as discussed in Headquarters Daily Report number H-99-0104. The evaluation written in the CARD determined that physical differences existed between EPR cables installed at Fermi and Davis-Besse and that the Fermi cables were highly resistant to water treeing. The CARD did not specify an action to establish tests to verify EPR cable integrity. No history of cable failures had occurred at Fermi from moisture intrusion.

In response to the inspectors' concern, the licensee documented the following proposed corrective actions in CARD 03-11668: 1) a solution team to develop and implement a cable aging program in support of future life extension; 2) evaluate Fermi current practices in condition monitoring low and medium voltage cables against practices recommended in Sandia report, SAND 96-0344; "Aging Management Guideline for Commercial Nuclear Power Plants -Electrical Cable Terminations;" 3) Develop a plan to support reliable cable performance and future life extension; 4) consider implementing a PM activity to inspect and drain any manholes containing water; and 5) review cable pulling compounds used at Fermi to determine if any potential adverse reaction between pulling compound and cable jacket exists.

Because the physical differences between the cables used at Fermi and at Davis-Besse differ significantly, the Fermi cables are rated for submergence, and that no relevant failures have occurred at Fermi, the inspectors concluded that there is adequate assurance that the cables will maintain their integrity until the licensee's corrective actions, as discussed above, have been fully implemented. This issue 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 October 3, 2003. The inspectors asked the licensee whether any material examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meetings

Interim exits were conducted for:

- Emergency Preparedness inspection with Mr. K. Morris on July 15, 2003.

- Public Radiation Safety radioactive waste processing and transportation programs inspection with Mr. H. Higgins on September 12, 2003. The inspector provided an inspection debrief to Mr. D. Cobb on September 12, 2003, prior to the formal interim exit meeting.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
D. Cobb, Plant Manager
H. Higgins, Radiation Protection Manager
R. Johnson, Licensing Supevisor
J. Korte, Nuclear Security Manager
K. Morris, Emergency Preparedness Supervisor
W. O'Connor, Jr., Vice President Nuclear Generation
N. Peterson, Nuclear Licensing Manager
S. Stasek, Nuclear Assessment Director

Nuclear Regulatory Commission
M. Ring, Chief, Division of Reactor Projects Branch 1

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000341/2003008-01	URI	Station Blackout Combustion Turbine Generator (CTG) Failure to Start During Loss of Electrical Grid (Section 1R14.1)
05000341/2003008-02	NCV	Failure to Apply Adequate Design Control Measures Results in Equipment Exceeding Its Design Basis Service Life (Section 4OA3.1)
05000341/2003008-03	NCV	Failure to implement design control processes for adding plastic sleeves on EDG drain line (Section 4OA5.1)

Closed

05000341/2003008-02	NCV	Failure to Apply Adequate Design Control Measures Results in Equipment Exceeding Its Design Basis Service Life (Section 4OA3.1)
05000341/2003002-01	URI	Fire on Emergency Diesel Generator (EDG) 11 Exhaust Pipe (Section 4OA5.1)
05000341/2003008-03	NCV	Failure to implement design control processes for adding plastic sleeves on EDG drain line (Section 4OA5.1)
05000341/2002011-01	URI	Low Flow Conditions On the Division 2 Emergency Equipment Cooling Water (EECW) System (Section 4OA5.2)
05000341/2003002-02	URI	Testing of Buried Ethylene Propylene Rubber (EPR) Cables (Section 4OA5.3)

LIST OF DOCUMENTS REVIEWED

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

1R01 Adverse Weather

System Operating Procedure 23.416, “General Service Water Pump House Heating and Ventilation,” Revision 9

Procedure 27.000.04, “Removal of Freeze Protection Measures,” dated 5/14/03

Warm and Cold Weather Preparation Work Update, dated 7/6/03

Procedure 27.000.04, Attachment 6, “Freeze Protection Measures,” Revision 25

Procedure 23.014, Revision 63

Performance Evaluation Procedure 27.000.06, “Hot Weather Operations,” Revision 0

Performance Evaluation Procedure 27.000.04, “Freeze Protection Lineup Verification,” Revision 26

Confined Space Permit for CST, “General Inspections,” dated 8/24/02

Confined Space Permit for CRT, “Monthly Freeze Protection,” dated 3/11/02

Confined Space Permit for CRT Pit, “Troubleshoot P1100-F611,” dated 7/8/03

Confined Space Permit for CRT Pit, “Maintenance on Pit Heather,” dated 6/26/03

Performance Evaluation Procedure, “Operator Rounds,” Revision 24

Job No. AD96030530, “Perform 27.000.06, Attachment 3, “Hot Weather System Readiness Review Checklist”

CARD 03-10811, “Incorrect Information on Procedure 27.322 Mayfly Infestation,” dated 6/25/03

CARD 02-16765, “Potential Warm Weather Preparation Improvements,” dated 6/24/02

Performance Evaluation Procedure 27.322, “Mayfly Infestation Preparation Plan,” Revision 1

CARD 03-17995, “RHR Complex EDG-13 HVAC Switchgear Room Return Air Damper

Actuator Leaking Oil," dated 7/2/03

CARD 03-17998, "EDG 11 Room Temperature Control Loop Malfunction," dated 7/2/03

CARD 03-17957, "RHR Complex EDG 13 HVAC Exhaust Air Gravity Dampers not Appearing to Work Properly," dated 6/19/03

CARD 03-1339, "A/C Units not Working," dated 7/2/03

CARD 03-17877, "Abnormal Noise From W4100C002," dated 5/29/03

CARD 01-13742, "Pneumatic Controller Output Low," dated 4/24/01

PEP 1.4.13, "Maintenance Rule Scope Determination for SSCS," Revision 0

1R04 Equipment Alignment

Job ID 000Z991930, Perform ASME Relief Valve Inspection in RF09

WR A409030100, Stanby Liquid Control (SLC) Pump C4103C001B Discharge Pressure Relief Valve, Dated 4/6/03

WR A407030100, Standby Liquid Control (SLC) Pump C4103C001B Discharge Pressure Relief Valve, Dated 4/6/03

Design Basis Document No. C41-00, Rev. 0, Standby Liquid Control System

Technical Specification 3.1.7, Standby Liquid Control (SLC) System

ARP 3D3, Rev. 8, SLC Tank Level High/Low

VTM VMR1-5, Rev. E, GE Type HFA51 and HFA151 Multicontact Auxiliary Relays

VTM VMR2-1, Rev. A, Union Pump Co. Type TD-60 Standby Liquid Control Pump

Procedure 24.139.02, Rev. 36, SLC Pump and Check Valve Operability Test

ST-OP-315-0014-001, Rev. 14, Operations Training Manual - Standby Liquid Control

Drawing 6M721-2082, Rev. AA, Diagram Standby Liquid Control System

Drawing 6I721-2131-02, Rev. H, Schematic Diagram Standby Liquid Control System Tank Heaters

Diagram 6I721-2131-01, Rev. J, Schematic Diagram Standby Liquid Control Pumps

GE Purchase Specification No. 21A9342AB, Standby Liquid Control System Pump Data Sheet, Dated 2/15/70

VTM VMC2-45, Rev. C, Conax Non-O-Cap Explosive Actuated valve Assembly

CARD 03-21159, Reinstall Insulation Around valve and Piping, Dated 8/12/03

CARD 03-19837, ARP 3D7 Needs Revised, Dated 8/11/03

Procedure 23.139, Rev. 36, Standby Liquid Control System

Procedure 29.ESP.05, Rev. 7, RPV Injection Using SLC Test Tank

Procedure 35.139.002, Standby Liquid Control (SLC) System Explosive Valve Insert Replacement

UFSAR 4.5.2.4, Standby Liquid Control System

WR No. C156010100, Replace SQUIB valve trigger assembly after detonation by surveil. 24.139.03

WR No. C145010200, Replace SQUIB valve trigger assembly after detonation by surveil. 24.139.03

UFSAR 7.4.1.2, Standby Liquid Control System Instrumentation and Control

UFSAR Figure 4.5-18, Sodium Pentaborate Volume Concentration Requirements

UFSAR Figure 4.5-17, Standby Liquid Control System PI&D

Drawing 6M721-5704, Rev. I, Standby Liquid Control System Functional Operating Sketch

Procedure 43.000.002, Rev. 33, ASME Section XI Relief Valve Setpoint Testing

Specification 3071-507, System Design Specification for Standby Liquid Control System, Dated 12/23/98

ARP 3D7, Rev. 8, SLC Tank Temperature High/Low

ARP 3D11, Rev. 7, SLC Ignition Continuity Loss

Procedure 74.000.19, Standby Liquid Control Sodium Pentaborate Surveillance - Monthly

SLC Tank sample data from 1/23/03 through 8/7/03

Completed 24.000.02, Shiftly/Daily - Mode 1, 2, 3 - Plant, Week of 7/27/03

Procedure 24.139.03, Rev. 38, SLC manual Initiation, RWCU Isolation, and Storage Tank Heater Operability Test

Procedure 29.ESP.02, Rev. 10, Alternate boron Injection

Standby Liquid Control System Health Report for 4th Quarter, 2002

DECO File No. R4-357, Certified Pump Performance Data, Serial Nos. 284200 (SLC Pump “A”) and 284201 (SLC Pump “B”), Dated 11/23/71.

1R05 Fire Protection

Drawing 4A721-4300, “Switchgear Room South Wall,” dated 3/1/99

UFSAR 9A.4.2.5, “Switchgear Room, Zone 4, El. 613 ft 8-1/2 inches

UFSAR, Figure 9A-6, “Fire Protection Evaluation Reactor and Auxiliary Building, Second Floor Plan (Elevation 613.5 ft), Revision 12

UFSAR, Figure 9A-8, “Fire Protection Evaluation Reactor and Auxiliary Building, Third Floor Plan, Elevation 641.5 ft and 643.5 ft,” Revision 12

USAR 9A.4.1.7, EECW Room, Revision 10

Reactor and Auxiliary Buildings Second Floor - Elevation 613' 6" Drawing No. 6A721-2405, Revision 10

UFSAR 9A.4.5, Turbine Building, Revision 7

UFSAR 9A.4.2.4, Relay Room, Zone 3, Revision 11

Reactor and Auxiliary Buildings Cable Tray Area Plan - Elevation 603' 6" Drawing No. 9A-5, Revision 11

UFSAR 9A.4.1.9, Reactor Building Fourth Floor, Zone 8, Revision 8

Reactor Building and Auxiliary Buildings - Elevation 659' 6" Drawing No. 9A-9, Revision 10

UFSAR 9A.4.8, General Service Water Pump House, Revision 8

CARD 03-19550, P7300F109B Has a Small Hydrogen Leak (Failed PMT), July 26, 2003

VTM VMB9-14.1, Ruskin Model CD31 and CDRI92 Control & Isolation Dampers, Rev.
A.

UFSAR (A.4.3, Residual Heat Removal Complex

UFSAR Figure 9A-15, Fire Protection Evaluation Residual Heat Removal Complex Upper Floor Plan

UFSAR Figure 9A-14, Fire Protection Evaluation Residual Heat Removal Complex

Grade Floor Plan

UFSAR Figure 9A-1, Fire Protection Evaluation Plot Plan

UFSAR Figure 9A-10, Fire Protection Evaluation Reactor and Auxiliary Buildings Fifth Floor Plan

UFSAR 9A.4.2.15, Control Room Ventillation Equipment Room and Standby Gas Treatment Rooms, Zone 14

Procedure 27.000.014, Enclosure A, “Aiming Criteria Battery Operated Emergency Lighting”

Plant Technical Procedure 28.502.07, CO_2 Fire Suppression Function Test Zone 14, Auxiliary Building, 3rd Floor,” dated 5/7/01

WR 000Z020357, “Revise PMT to Reflect Work Actual Steps Required,” dated 5/16/02

Plant Technical Procedure 28.507.04, Revision 4, “Test and Inspection of Fire Dampers,” dated 8/14/02

UFSAR, Figure 9A08, Revision 12, “Fire Protection Evaluation Reactor and Auxiliary Buildings 3rd Floor Plan Elevation 641.5ft and 643.5 ft”

UFSAR 9a.4.2.11, “Divisions I and II Battery Rooms, Zone 10, El. 643 ft. 6 in.”

Plant Technical Procedure 28.507.05, Revision 10, “Inspection of Penetration Fire Stops,” dated 11/4/02

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

SS-OP-904-1056, “Seismic Event/RCIC Spurious Initiation w/Failure to Trip/Spurious Trip of 65E-E6/LOCA,” Revision 1

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

Feedwater/RHR Injection Check Valves Get Well Plan, dated 2/4/2002

Maintenance Rule Conduct Manual, dated 5/3/2002

Program Health Reports - Primary Containment, 3rd and 4th Quarters 2002

Program Health Reports - High Pressure Coolant Injection, 3rd and 4th Quarters 2002

Program Health Reports - Residual Heat Removal, 3rd and 4th Quarters 2002

CARD 01-20113, Check Valve C4100F006 failed LLRT, dated 11/15/2001

CARD 02-11585, Potential ‘Jack Rabbit’ Start of HPCI, dated 1/12/2002

CARD 02-16367, E41F401 Failed Stroke Time During Testing, dated 11/7/2002

CARD 02-18678, Loose Wiring Found in the E1150-F068A Open Ckt, dated 9/6/2002

CARD 03-16252, ASME Relief Valve would not Lift at 110% of Set Pressure, dated 4/8/2003

CARD 03-18539, Inadequate Maintenance Rule Monitoring, dated 7/11/2003

1R13 Maintenance Risk Assessment and Emergent Work

Plan of the Day, dated 8/5/03

Weld Process Control Sheet No. 000Z033202 for PIS No. G3300F120, Dated 8/28/03

Welding Specification procedure No. A11-3.1, Rev. 1, Dated 3/15/84

Fermi Welding manual, WPS/BPS Index/Matrix, Rev. 15

Memo from S. G. Garreffa & J. Polacheck to L. Burkholder, Dated 9/4/03, Re: Welder/Brazer Qualification Log.

WR No. 000Z033202, Body to Bonnet leak on G3300F120 Seal Weld per Replacement Installation Document 72424

Detroit Edison Log. No. 03-035, Rev. 1, “Detroit Edison Co. ASME Section XI Repair and Replacement program for Fermi 2 Power Plant,” Dated 8/28/03.

Replacement Installation Document 72424, Rev. 0, Dated 8/28/03.

1R14 Nonroutine Plant Evolutions

CARD 03-19464, “Inverter Failure du to Loss of Electrical Grid,” dated 8/14/03

WR 000Z033553, “Invest, TS cause of Inverter failure and Inspect MCC CTG-1 Pos. BB-4 DC Fuel F P,” dated 8/4/03

WR 000Z033258, “Install TSR 32666, reset CTG 11-1's Inverter Low DC Low Volt Shutdown Setpoint,” dated 9/4/03

Technical Service Request 32666, “Low Voltage Setpoint Change for CTG 11-1,” dated 8/16/03

1R15 Operability Evaluations

Engineering Functional Analysis, System J1100, dated 6/23/03

CARD 03-18619, “Potentially Inadequate LHGR Thermal Limit for Single Loop Operation in Cycle 10, dated 6/13/03

Engineering Function Analysis, System E5100 RCIC, dated 6/30/03

Engineering Functional Analysis, System P4400 EECW, dated 11/25/03.

ISI/NDE-IST-Program Evaluation Sheet Log No. 03-0037, Revision 0, “G3300F120 Operational Mechanical Joint Leakage Evaluation”

ISI/NDE-IST-Program Evaluation Sheet Log No. 03-0037, Revision 1, “G3300F120 Operational Mechanical Joint Leakage Evaluation”

G3300F120 Leakage Monitoring Plan

Equivalent Replacement Evaluation, RID 72424, Revision 0

Drawing 6M721-2336-1, Revision Y, “Piping Plant & Elevation Feedwater System Inside Drywell Reactor Building”

Drawing 4M721-4574, Revision G, “LLRT Penetration X-9B Feedwater System Reactor Building”

Drawing 4M721-3245-1, Revision Z, “Piping-Isometric Reactor Water Cleanup Pump Discharge”

Drawing, “4" - 900 Weld Ends Carbon Steel Pneumatic Cylinder Spring Assist Close Swing Check Valve”

Summary of Changes, Tests, and Experiments for Grand Gulf Nuclear Station from 9/16/00 through 4/30/91

<u>1R17</u> <u>Permanent Plant Modifications</u>

EDG 30251, for upgrading the integrated plant computer system

<u>1R19</u> <u>Post Maintenance Testing</u>

Procedure 64.080.438, Reactor Building Ventilation Exhaust Radiation Monitor, Division 2 Calibration, Revision 8

Procedure 46.626.004, Gulf Atomic RP-30/RP-30A Process Radiation Monitor Module, Revision 24

<u>1R22</u> <u>Surveillance Testing</u>

CARD 03-11668, “NRC Question Regarding Testing of Underground Cables,” dated 3/3/03

CARD 99-18796, “OIM 99-152; MR No. H-99-0104 - Failure of 5KV Cable in Underground Conduit,” dated 12/17/99

Meeting Minutes for Design Spec 3071-080, dated 6/9/03

Safety Guide 30, “Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment,” dated 8/11/72

IEEE Standard 336-1985, “IEEE Standard Installation, Inspection, and Testing Requirements for Power, Instrumentation, and Control Equipment at Nuclear Facilities

Design Specification 3071-080, “Detroit Edison Design and Procurement Specification for Special Wires and Cables used at Fermi 2,” dated 6/25/02

LER 05000255/96-002-01, “Initiation of Technical Specifications Required Shutdown Due to Safeguards Cable Fault - Supplemental Report,” report date 10/4/96

LER 05000275/93-005-01, “Medium Voltage Cable Failures Due to Chemical Degradation and Unknown Causes,” report date 10/4/93

National Fire Code, Chapter 10, “Power Cables”

Plant Technical Procedure 24.307.36, Revision 42, “DGSW, DFOT and Starting Air Operability Test - EDG 13

Temporary Change Notice 11031, “Provided Clarification of I&C Portion of DGSW Procedure for Isolation and Blowdown of Flow Instrument Lines” dated 2/10/03

WR W482030100, “Recal EDG 13 DGSW Pump ‘B’ Low Flow Switch,” dated 8/13/03

1R23 Temporary Plant Modifications

Temporary Modification 03-0020, “Removal From Service of Div. 1 Safety Relief Valve B2104F013A Tailpipe Pressure Switch B21N411A”

Job 000Z022359, “Install Temporary Mod 02-0020”

1EP4 Emergency Action Level and Emergency Plan Changes

______Fermi Nuclear Power Plant Emergency Plan; Revisions 26, 27, and 28

1EP6 Drill Evaluation

Fermi 2 Team List; July 2, 2003; Rev. 17.

Scenario 33A, Drill Package.

Fermi 2 RERP Team List, dated 7/2/03

2PS2 Radioactive Material Processing and Transportation

Certificate of Qualification: Hazardous Materials Worker - Level 2; dated March 11, 2002

Qualification Matrix: Hazardous Materials Worker - Level 1; dated September 3, 2003

Abnormal Operating Procedure 20.000.27; Transportation Accidents Involving Radioactive Material from Fermi 2; Revision 7

Audit Report 02-0112; Nuclear Quality Assurance Audit Report: Radiation Protection, Radioactive Effluent Monitoring, Radiological Material, Transfer and Disposal, and Non-Radiological Environmental Protection Programs; dated October 7 through November 25, 2002

CARD 02-19701; Evaluate Vacuum Discrepancy in RDS-1000 Operating Procedure vs. Topical Report Identified During RW Self Assessment; dated October 4, 2002

CARD 02-19702; Observations on Procedure Compliance during Radwaste Self Assessment; dated October 4, 2002

CARD 02-19706; Description in UFSAR for RDS-1000 Dewatering Equipment is Not Correct Since Detroit Edison Purchased Equipment; dated December 16, 2002

CARD 02-21296; Assess Method of Evaluating Process Control Program; dated December 20, 2002

CARD 03-21697; NRC Observation During Review of Radioactive Shipment Documentation; dated September 11, 2003 (NRC-identified issue)

LP-GN-528-0003; Hazardous Material (HAZMAT) Orientation, Function Specific Training - Level 1; Revision 3

MGA20; Transportation Security Plan; Revision 0

MRP19; Shipping Notifications; Revision 6

MPR21; Radwaste Shipping Operations; Revision 9

MRP24; Fermi 2 10CFR61 Compliance Manual; Revision 2

MRP26; Process Control Program; Revision 0

NRC-03-0043; Annual Radioactive Effluent Release and Radiological Environmental Operating Reports; dated April 29, 2003

NPRC-02-0173; Scaling Factors Report Dated April 25, 2002; dated May 30, 2002

NPRC-02-0183; Validation of Stainless Steel Laundry Container Shipment Using DAW Scaling Factors, Sample Reference Date - 2/26/2002; dated June 6, 2002

NPRC-02-0377; Focused Self-Assessment of Radwaste Processing with the RDS-1000; dated December 18, 2002

NPRC-03-0193; Scaling Factors Report Dated January 14, 2003; dated July 31, 2003

NPRC-03-0208; Transportation Security Risk Assessment (CONFIDENTIAL - NOT FOR PUBLIC DISCLOSURE); dated August 19, 2003

NPRC-03-0226; Scaling Factors Report Dated August 26, 2003; dated September 5, 2003

PTP 65.000.506; Shipping Low Specific Activity (LSA) Radioactive Material; Revision 16

PTP 65.000.522; Shipping Surface Contaminated Object Radioactive Material; Revision 4

PTP 65.000.523; Radwaste Shipments; Revision 6

PTP 65.704.001; Setup and Operating Procedure for the RDS-1000 Unit; Revision 0

PTP FO-OP-032-483; Set Up and Operating Procedure for the RDS-1000 Unit at Detroit Edison Fermi-2; Revision 24

Radioactive Material Shipment 02-005; Dewatered Powdered Resin LH-01-004; dated January 25, 2002

Radioactive Material Shipment 02-023; 10 CFR 61 Samples for Analysis; dated March 20, 2002

Radioactive Material Shipment 02-036; Dewatered Powdered Resin LH-02-002; dated May 23, 2002

Radioactive Material Shipment 02-061; CD-600 Dryer and Other Equipment; dated October 28, 2002

Radioactive Material Shipment 03-030; Contaminated Laundry; dated April 5, 2003

Radioactive Material Shipment 03-033; SRV and SRV Pilots; dated April 11, 2003

Radioactive Material Shipment 03-058; Dewatered Powdered Resin LH-03-001; dated June 11, 2003

Radioactive Material Shipment 03-068; High Rad Dry Active Waste (DAW); dated July 9, 2003

UFSAR Chapter 11.5; Solid Radwaste System; through Revision 9

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

Dose Equivalent Iodine Data Spreadsheet (maintained by Chemistry); dated January 2002 through June 2003

Liquid and Gaseous Effluent Data/Calculations; dated 4th Quarter 2002 through 2nd Quarter 2003

CHS-PRI-06; Chemistry Gamma Spectroscopy Analysis Report, DEI SP-50; dated September 10, 2003

NRC-03-0043; Annual Radioactive Effluent Release and Radiological Environmental Operating Reports; dated April 29, 2003

PTP 76.000.34; Reactor Coolant Analysis; Revision 10

4OA2 Identification and Resolution of Problems (71152)

CARD 03-12851, "Unexpected Response While Reducing RCIC Turbine Speed," dated 3/11/03

CARD 02-10885, "E5150F044 Failed to Open at Least 80 percent Without Assistance," dated 1/27/02

CARD 01-22421, "RCIC Control Valve Fails Weekly Performance Test," dated 12/16/01

CARD 00-19367, "E5150-F944 Hard to Operate," dated 9/10/00

CARD 99-02412, "RCIC Control Valve Not Going Full Open," dated 11/7/99

4OA3 Event Followup (71153)

CARD 03-00518, "Fire at EDG 11, Root Cause Analysis Report," dated 6/24/03

Fermi Business Practice 39, Rev. 2 (including Appendices A-G).

Job ID 0256030818, Core Spray System discharge piping filled and valve position verification, dated 8/18/03.

General Information Lead Sheet Environmental Qualification Central File, EQ0-EF2-018, Approved 8/31/99

DER 93-0608, "OTH 93-051: OE 6263: Premature Agastat Relay Failure," dated 10/25/93

EF2-49969, "10CFR50.55(e) Report on Agastat Relay Bases (#23)", dated 4/30/81.

EF2-49784, "10CFR50.55(e) Report on Agastat Bases", dated 10/7/80.

NRC-93-0103, "Response to Inspector Followup Item 93-010-03," dated 9/1/93

NEDO-32291-A, Supplement 1, "System Analyses for the Elimination of Selected Response Time Testing Requirements," dated 10/99

Fermi 2 License Amendment 151, dated 10/02/02

CARD 03-19313, “Review NRC Report 50-416/03-06, Specifically the Discussions on Pages 6, 7, 10 and 14 in Regards to the Establishment and use of PM activity ‘Grace’ periods at Fermi,” dated 8/1/03

EEQ Lead Sheet, EQ!-EF2-048, Power Relays, Revision C

Detroit Edision Memo NE-NS-87-0075, “Service Life of Relays in Safety-Related Systems,” dated 3/10/87

DER 92-0492, “Missed Preventive Maintenance on A71 Agastat Relays,” dated 10/31/92

Schematic Diagram 6I721-2155-15, “Reactor Protection System Testability Modification, Revision I

Schematic Diagram 6I721-2155-16, “Reactor Protection System Testability Modification, Revision H

Schematic Diagram 6I721-2155-065, “Reactor Protection System Trip System ‘A’ System Relays, Revision R

Schematic Diagram 6I721-2155-08A, “Reactor Protection System Trip System ‘A2' SCRAM Trip Logic, Revision A

CARD 03-10985, “Agastat Relay Replacement Exceeded Time in GE SIL/NRC Information Notice,” dated 7/19/03

Detroit Edison Letter NRC-02-0036, “Proposed Technical Specification Change (License Amendment) Response Time Testing,” dated 5/23/02

Environmental Qualification Assessment, Qualified Life Evaluation for Agastat Relays (Normally De-Energized), Revision 0

DER 87-566, “PM’s not Completed as Scheduled,” dated 1/5/88

Detroit Edison Memo NANL 02-0088, “RACTS Weekly Management Report, “ dated 11/15/02

CARD 99-13238, “OTH 99-046: OE 9785 Qualified Service Life for Agastat Relays,” dated 4/15/99

Maintenance Procedure 35.218.017, “Inspection and Testing of Multi-Contact Auxiliary Relays,” Revision 37

Surveillance Procedure 44.010.001, “RPS - Reactor Steam Dome Pressure, Division 1, Functional Test,” Revision 28

CARD 03-18478, “Wrong Relay Replaced,” dated 6/27/03

Operator Logs from 6/27/03 - 6/28/03

Operator Logs from 6/30/03 - 7/2/03

Internal-External Wiring Diagram 6I721-2282-58, “RPS ‘A2' Trip Unit Cabinet H21-P085,” Revision E

Diagram 6I721-2281-35, “RPS ‘A2' Trip Unit Cabinet H21P085,” Revision I

Internal-External Wiring Diagram 6I721-2282-53, “RPS ‘A1' Trip Unit Cabinet H21-P085,” Revision F

Schematic Diagram 6I721-2155-16, “RPS Testability Modification,” Revision H

Schematic Diagram 6I721-2155-06, “RPS Trip System ‘A’ System Relays,” Revision R

WR C010970214, “Miscellaneous Signal Conditioner Reactor Pressure SCRAM Trip Unit Division 2,” dated 12/29/97

WR C010980129, “RPS Trip Channel ‘A2' Reactor Vessel High Steam Line Pressure K206C Relay,” dated 3/25/03

WR C013970826, “RPS Trip Channel ‘A2' Reactor Vesssel Low Water Level 2 SCRAM K208C Relay, dated 10/31/02

Arrhenius Calculation for Service Life on C7100M206C

RACTS 93239, “IFI 93-010-03 Agastat GP Series Relay not in the PM Program”

Jobs C010980129 and C010970214, “Replace RPS Channel ‘A2' Reactor Vessel High Steam Line Pressure Agastat Relay”

Job 0004030624, “Perform 44.010.0001 RPS-Reactor Steam Dome Pressure Division 1 Functional Test”

Job 0704030328, “Perform 44.010.007 RPS-Reactor Steam Dome Pressure Trip System A Channel A2 C Cal”

Job 000Z032679, “Wrong Relay Replaced Reinstall Correct Relay”

E1102C002A (RHR Pump “A”) performance plot from October 18, 2001 to August 1, 2003

Job ID 0266030318, Perform 24.204.06 Div. 2 LPCI & Torus Cooling Spray Pump & Valve Oper Test, Dated 3/18/02

Job ID 0261030204, Perform 24.204.01 Div. 1 LPCI & Torus Cooling Spray Pump & Valve Oper Test, Dated 2/4/03

Job ID 0261030506, Perform 24.204.06 Div. 2 LPCI & Torus Cooling Spray Pump & Valve Oper Test, Dated 2/29/0

E1102C002B (RHR Pump "B") performance plot from September 1, 2001 to August 14, 2003

Job ID 0266030617, Perform 24.204.06 Div. 2 LPCI & Torus Cooling Spray Pump & Valve Oper Test, Dated 4/21/03

Job ID 0266030618, Perform 24.204.06 Div. 2 LPCI & Torus Cooling Spray Pump & Valve Oper Test, Dated 8/13/03

CARD 03-12037, "Sargent & Lundy Technical Alert TA 2003-0006: KITTY and STRCOIL Program Error Overestimates Room Cooler Performance at Low Water Flow Rates," dated 4/16/03

Detroit Edison Memo No. TMPE 03-0167, "CARD 02-16602 Past Operability Followup," dated 7/21/03

CARD 03-21303, "EECW Flow out of Spec Low to Division 2 Battery Charger Room Cooler," dated 8/21/03

Job No. AJ93021211, "Division 2 EECW Throttled Valve Flow Verification," dated 12/11/02

CARD 02-19698, "Evaluate Treatment Chemicals for RBCCW/EECW/SCS," dated 10/29/02

CARD 02-16602, "Flows to Division 2 EECW Throttled Loads Outside Required Bands," dated 12/11/02

Job No. AJ93011211, "Division 2 EECW Trhottled Valve Flow Verification, dated 12/11/02

CARD 19263, "EECW Flow to T4100B019 was Left Lower Than Design (Division 2 RHR Room Cooler)," dated 12/11/02

Sargent & Lundy Evaluation No. 2003-01660, Revision 0, "Analysis of LOCA Heat-up of Reactor Buidling with Reduced Division II EECW Flowrates to the Core Spray (T4100B020) and Thermal Recombiner (T4100B037) Room Coolers, dated 6/16/03

LIST OF ACRONYMS USED

CARD	Condition Assessment Resolution Document
cfm	Cubic Feet per Minute
CFR	Code of Federal Regulations
CTG	Combustion Turbine Generator

DAW	Dry Active Waste
DC	Design Calculation
DEI	Dose Equivalent Iodine
EDG	Emergency Diesel Generator
EECW	Emergency Equipment Circulating Water
EPR	Ethylene Propylene Rubber
gpm	gallons per minute
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LSA	Low Specific Activity
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
PI	Performance Indicator
PM	Preventive Maintenance
Radwaste	Radioactive Waste
RHR	Residual Heat Removal
SCO	Surface Contaminated Object
SDP	Significance Determination Process
SRV	Steam Relief Valve
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
WR	Work Request