September 5, 2001

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

SUBJECT: FERMI 2
NRC INSPECTION REPORT 50-341/01-12(DRP)

Dear Mr. O'Connor:

On August 10, 2001, the NRC completed an inspection at your Fermi 2 Nuclear Power Station. The enclosed report documents inspection findings which were discussed on August 10, 2001, with you 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. Specifically, this inspection focused on plant operations, heat sink performance, 10 CFR 50.59 program, and security access authorization.

Based upon the results of this inspection, a GREEN finding that was a violation of NRC requirements was identified. However, because of its very low safety significance and because it has been entered into your corrective action program, the NRC is treating this issue as a Non-Cited Violation (NCV), consistent with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this Non-Cited Violation, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspectors at the Fermi 2 Nuclear Power Station.

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/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 50-341/01-12(DRP)

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

Docket No:	50-341
License No:	DPR-43
Report No:	50-341/01-12(DRP)
Licensee:	Detroit Edison Company
Facility:	Enrico Fermi, Unit 2
Location:	6400 N. Dixie Hwy. Newport, MI 48166
Dates:	July 1 through August 10, 2001
Inspectors:	S. Campbell, Senior Resident Inspector J. Larizza, Resident Inspector P. Pelke, Reactor Engineer G. O'Dwyer, Reactor Inspector P. Lougheed, Reactor Inspector G. Hausman, Reactor Inspector J. Belanger, Senior Physical Security Inspector
Approved by:	Mark Ring, Chief Branch 1 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000341-01-12, on 7/01-8/10/01, Detroit Edison Company, Fermi 2 Nuclear Power Station. Heat sink performance inspection.

The inspection was conducted by resident and specialist inspectors. This inspection identified one Green issue which involved a Non-Cited Violation. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

Inspector Identified Finding

Cornerstone: Mitigating Systems

GREEN. One Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," was identified for the licensee's failure to appropriately incorporate into the residual heat removal (RHR) heat exchanger test acceptance criteria the requirements from applicable design documents.

The safety significance of this finding was very low because the affected mitigation system remained operable. This issue was considered more than minor, because if left uncorrected, it could impact the ability of the licensee to detect degradation or loss of RHR heat exchanger function (Section 1R07.b.1).

Report Details

1. REACTOR SAFETY

Plant Status

Fermi 2 operated at or near 100 percent power throughout the inspection period. On July 28, 2001, power was decreased to 75 percent to perform control rod pattern adjustments and other planned maintenance and surveillance activities. Reactor power was increased to 100 percent on July 29, 2001.

1R02 Evaluations of Changes, Tests, or Experiments (71111.02)

.1 Review of Evaluations and Screenings for Changes, Tests, or Experiments

a. Inspection Scope

The inspector reviewed eight safety evaluations performed pursuant to Title 10 Code of Federal Regulations (10 CFR) 50.59. The safety evaluations were related to temporary and permanent plant modifications, setpoint changes, procedure changes, potential conditions adverse to quality, and changes to the licensee's updated safety analysis report. The inspector confirmed that the safety evaluations were thorough and that prior NRC approval was obtained as appropriate. The inspector also reviewed 12 safety reviews (screenings) where the licensee had determined that a 10 CFR 50.59 safety evaluation was not necessary. In regard to the changes reviewed where no 10 CFR 50.59 safety evaluation was performed, the inspector verified that the changes did not meet the threshold to require a 10 CFR 50.59 safety evaluation. These safety evaluations and screenings were chosen based on risk significance of samples from the different cornerstones.

b. Findings

No findings of significance were identified.

.2 Identification and Resolution of Problems

a. Inspection Scope

The inspector reviewed the licensee's condition assessment resolution documents (CARDs) concerning 10 CFR 50.59 safety evaluations and screenings to verify that the licensee had an appropriate threshold for identifying issues. The inspector evaluated the effectiveness of the corrective actions for the identified issues.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04Q)

a. Inspection Scope

The inspectors used piping and instrumentation diagrams, surveillance procedures and condition assessment resolution documents (CARDs), to verify proper alignment of valves, test switches, control switches and cleared annunciator alarms for the following emergency power sources:

- Emergency Diesel Generator 13
- Emergency Diesel Generator 14

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

Quarterly Tour of Risk Significant Areas for Fire Protection

a. Inspection Scope

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

- High Pressure Coolant Injection Room and Control Rod Drive Mechanism Room (UFSAR Zone 3)
- Second Floor Reactor Building (UFSAR Zone 6)
- Fourth Floor Reactor Building (UFSAR Zone 8)

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

Biennial Review of Heat Sink Performance

a. Inspection Scope

Two specialist inspectors reviewed documents associated with testing, inspection, cleaning, and performance trending of the Residual Heat Removal North Heat Exchanger E1101B001A, Emergency Diesel Generator No. 11 Intake Air East Air Cooler R3001B029A, and Emergency Equipment Cooling Water Division 2 Heat Exchanger P4400B001B. These heat exchangers were chosen based upon their importance in supporting required safety functions as well as relatively high risk achievement worths in the plant specific risk assessment. The Residual Heat Removal North Heat Exchanger and Emergency Equipment Cooling Water Division 2 Heat Exchanger were also selected to evaluate the licensee's thermal performance testing methods. During the inspection, the inspectors reviewed completed surveillance tests and associated calculations, and performed independent calculations to verify that these activities adequately ensured proper heat transfer. The inspectors reviewed the documentation to confirm that the test or inspection methodology was consistent with accepted industry and scientific practices, based on review of heat transfer texts and Electrical Power Research Institute standards (EPRI NP-7552, Heat Exchanger Performance Monitoring Guidelines, December 1991 and EPRI TR-107397, Service Water Heat Exchanger Testing Guidelines, March 1998).

The inspectors reviewed condition reports concerning heat exchanger and ultimate heat sink performance issues to verify that the licensee had an appropriate threshold for identifying issues and entering them in the corrective action program. The inspectors also evaluated the effectiveness of the corrective actions for identified issues, including the engineering justification for operability, if applicable.

The documents that were reviewed are included at the end of the report.

b. Findings

Residual Heat Removal Heat Exchanger Testing Deficiencies

GREEN. One Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” was identified for failure to appropriately incorporate into the test acceptance criteria the requirements from applicable design documents. During the inspection the inspector reviewed the Residual Heat Removal (RHR) test procedures (see list of documents) and identified the following nonconservative deficiencies:

- Since Revision 6, dated September 3, 1998, the tests used an incorrect value of 117 million British Thermal Units per hour (BTU/hr) as the required design basis heat load. This value had been obtained from a TEMA Data Sheet from the vendor, Fromson Heat Transfer, LTD. However, the correct design heat load was 127.4 million BTU/hr as required by the accident analysis. The inputs to the accident analysis were documented in UFSAR, Revision 10, Table 6.2-1, “Containment Parameters.” The licensee initiated Condition Assessment Resolution Document (CARD) 01-13240 in response to this deficiency. The licensee also had no formal calculation to translate the applicable design requirements into the test procedures' acceptance criteria. One of the corrective actions specified by the CARD was the creation of this calculation.

- The tests used an incorrect minimum design RHR Service Water flow of 9,000 gpm. However, in 1995 to allow for RHR Service Water pumps’ degradation, the licensee reduced the minimum design RHR Service Water flow to 8,250 gpm. The licensee initiated CARD 01-13241 to address this concern.

- Also, a modeling uncertainty was not quantitatively included in the tests' uncertainty acceptance criteria. The licensee initiated CARD 01-13239 to address this concern.

The licensee's and the inspector's preliminary calculations indicated that both of the RHR heat exchangers' actual performances were still well above design requirements so the inspector had no operability concerns. This finding was more than minor because if left uncorrected a heat exchanger that did not meet design requirements could have gone undetected. However, since only the mitigation cornerstone was affected and the system remained operable, the finding screened by the SDP, is considered to be of very low safety significance (Green). 10 CFR Part 50, Appendix B, Criterion XI, "Test Control" requires that requirements from applicable design documents be incorporated into test procedures. Contrary to 10 CFR Part 50, Appendix B, Criterion XI, the licensee failed to appropriately incorporate into the test acceptance criteria the requirements from applicable design documents and this is an apparent violation. However, due to the very low safety significance of the items and because the licensee entered these items into the corrective action program, this violation is being treated as a Non-Cited Violation (**NCV 50-342/01-12-01**) in accordance with Section VI.A.1 of the NRC Enforcement Policy.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

On July 25, 2001, the inspectors observed licensed operator performance in mitigating the consequences of events at the simulator training facility as well as the licensed operator self and evaluator critique of the just completed high risk actions to mitigate these events. The inspectors observed crew performance, ability to take timely actions, verification of alarms, correct usage and implementation of abnormal and emergency operating procedures, oversight and direction provided by the shift manager, the ability to implement appropriate Technical Specification and emergency plan actions and notifications.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12Q)

a. Inspection Scope

The inspectors reviewed system health reports, associated CARDs, white papers for probabilistic risk assessment on conditional probabilities and the control room unit logs for Reactor Controls, C1100, to determine whether the maintenance rule program had been implemented appropriately by assessing the characterization of failed structures, systems, and components. The inspectors also determined whether goal setting and performance monitoring were adequate.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed the following conditions to determine whether emergent work activities were performed in a manner that did not place the plant in an unacceptable configuration, and to verify that the licensee managed plant risk adequately:

- Trip of the CM and CT breakers causing the 345 KV Brownstown 3 line to de-energize.
- Emergency Diesel Generator 14 switchgear room NCX relay failed to seal-in during surveillance 42.307.02.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

.1 Lack of Oil Indication in Standby Feed Water Pump "A" Outboard Motor Bearing

a. Inspection Scope

On July 26, 2001, the inspectors reviewed oil levels for risk significant and safety-related equipment as part of the supplemental inspection (Inspection Report 50-341/01-010) for the Emergency Alternating Current Performance Indicator crossing from the GREEN to WHITE threshold. The Emergency Alternating Current Performance Indicator crossed the threshold when fault exposure hours were accumulated when the alternator outboard bearing for emergency diesel generator 14 catastrophically failed about twelve hours into a twenty-four-hour endurance run. An inadequate oil level in the bearing caused the failure. The review included the standby feedwater pumps "A" and "B" motor and pump oil levels. The inspectors interviewed operations and maintenance personnel and reviewed maintenance and operations procedures.

b. Findings

The inspectors identified that the outboard motor bearing oil level indicator for standby feed water pump "A" did not show an oil level. A leaking auxiliary oil pump discharge check valve or relief valve may be causing the slow draining of the outboard reservoir. Because this condition was identified one day before completing the supplemental inspection, the inspectors addressed the condition in this report. The licensee wrote Level 3 CARD 01-17205 and initiated a tracking limiting condition for operations (LCO) to evaluate pump operability with this deficient condition.

A 15 minute run was performed, vibration measurements were taken, and the bearing oil was sampled. No degradation was noted. During the test, the licensee manually operated the auxiliary oil pump and timed filling of the bearings. The inboard bearing took five seconds to fill and the outboard took three minutes. CARD 01-17269 was written to document that the inboard orifice had excessive flow. Because the inboard and outboard bearings are supplied from the auxiliary oil pump discharge pipe, the excessive flow to the inboard bearing could lessen the oil flow to the outboard bearing. An orifice on the inboard oil supply pipe may be missing. The licensee's evaluation (MES 27) addressed excessive flow and documented that the pump remained operable if oil were in the sight glass. Operators restored oil level and increased the frequency of verifying the oil levels.

The inspectors identified the inadequate oil level because a check of these levels was not in the operators' rounds log. Further, through interviews, the inspectors discovered that the operators did not check the motor bearings for level before starting the pump manually during routine testing, missing another opportunity to identify this condition. Also, the extent-of-condition review and corrective actions for CARD 01-14004, which documented the emergency diesel generator 14 outboard bearing failure, may have been narrowly focused since this pump was not included in the scope for an extent of condition review.

At the close of this inspection, the licensee had not determined the root causes for the slow draining of the outboard bearing oil levels and the excessive flow to the inboard bearing. Also, the licensee had not completed a past operability assessment of the pump with the degraded condition. The inspectors considered this an **Unresolved Item (URI 50-341/01-012-02)** pending the root cause determinations for these issues and the inspectors' review of the associated corrective actions.

.2 Engineering Functional Analysis for Operability Determination

a. Inspection Scope

The inspectors reviewed equipment evaluations to determine if operability was properly justified and the component or system remained available such that no unrecognized increase in risk occurred. Evaluations for the following equipment issues that occurred during the inspection were reviewed:

- Emergency Diesel Generator broken studs on the vertical drive component.
- Electrical loading on Combustion Gas Turbine 11-1 for Appendix "R" dedicated shutdown scenario.
- Leakage from Emergency Equipment Closed Cooling system valve P4400-F129A, "Division 1 EECW return to heat exchanger inlet isolation valve leak-off nipple."

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed post maintenance testing packages to confirm that the tests were adequate for the scope of the maintenance. The inspectors also determined that the tests restored the operational readiness consistent with the design and licensing basis documents.

- Emergency Diesel Generator 11 slow start and load test.
- Limitorque Motor Operator periodic inspection procedure following replacement of MOV fasteners with the valve closed (stem under compression).

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors witnessed and reviewed test data for the following surveillance tests. The inspectors reviewed Technical Specifications to confirm that surveillance activities had verified the equipment performed the intended safety functions and met operational readiness. The inspectors verified sufficient staffing levels of the control room and other personnel to adequately conduct the test.

- Emergency Core Cooling System Reactor Recirculation Riser Differential Pressure.
- Emergency Core Cooling System, Core Spray System, Logic Functional.
- Division 2 Emergency Diesels Generators Emergency Core Cooling System Start Circuits.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed a temporary modification where a Rochester Instrument System (RIS) microalarm annuciator was installed to monitor the 345 KV circuit breaker compressor trouble alarm contacts. The inspectors reviewed this installation to determine whether it may result in a departure from the design basis and success criteria to mitigating core damage and to determine if the availability, reliability or functional capability of the system was impacted.

b. Findings

No findings of significance were identified.

3. SAFEGUARDS

Physical Protection (PP)

3PP1 Access Authorization (AA) Program (Behavior Observation Only) (71130-01)

a. Inspection Scope

The inspector interviewed five supervisors and five non-supervisors (both licensee and contractor employees) to determine their knowledge level and practice of implementing the licensee's behavior observation program responsibilities. Selected procedures pertaining to the Behavior Observation Program and associated training activities were also reviewed. Also licensee fitness-for-duty semi-annual test results were reviewed. In addition, the inspectors reviewed a sample of licensee self-assessments, audits, and security logged events. The inspectors also interviewed security supervisors to evaluate their knowledge and use of the licensee's corrective action system.

b. Findings

No findings of significance were identified.

3PP2 Access Control (Identification, Authorization and Search of Personnel, Packages, and Vehicles) (71130.02)

a. Inspection Scope

The inspector reviewed the licensee's protected area access control testing and maintenance procedures. The inspector observed licensee testing of all access control equipment to determine if testing and maintenance practices were performance based. On two occasions, during peak ingress periods, the inspector observed in-processing search of personnel, packages, and vehicles to determine if search practices were conducted in accordance with regulatory requirements. Interviews were conducted and records were reviewed to verify that security staffing levels were consistently and appropriately implemented. Also the inspector reviewed the licensee's process for limiting access to only authorized personnel to the protected area and vital equipment by a sample review of quarterly access authorization reviews performed by managers. The inspector reviewed the licensee's program to control hard-keys and computer input of security-related personnel data.

The inspector reviewed a sample of licensee self-assessments, audits, maintenance request records, and security logged events for identification and resolution of problems. In addition, the inspector interviewed security managers to evaluate their knowledge and use of the licensee's corrective action system.

b. Findings

No findings of significance were identified.

3PP4 Security Plan Changes (71130.04)

a. Inspection Scope

The inspector reviewed Revision 37 to the Fermi 2 Nuclear Plant Security Plan to verify that the changes did not decrease the effectiveness of the submitted document. The referenced revision was submitted in accordance with 10 CFR 50.54(p) by a licensee letter dated May 24, 2001.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA2 Performance Indicator Verification (71151)

.1 Reactor Coolant System Specific Activity Performance Indicator Verification

a. Inspection Scope

The inspector reviewed the reactor coolant system specific activity analysis data from April 1, 2000, to March 31, 2001. The inspector verified that the data reported by the licensee was accurate and complete. On August 1, 2001, inspectors observed a chemistry technician obtain and analyze a reactor coolant sample for Dose Equivalent Iodine (131) DEI.

b. Findings

No findings of significance were identified.

.2 Reactor Coolant System Leak Rate Performance Indicator Verification

a. Inspection Scope

The inspectors reviewed reactor coolant system leak rate data for April 1, 2000, to March 31, 2001. The inspectors verified that the data reported by the licensee was accurate and complete. The inspector observed an operator calculate the identified leakage rate on July 25, 2001.

b. Findings

No findings of significance were identified.

.3 Mitigating System and Initiating Events Performance Indicator Verification

a. Inspection Scope

The inspector reviewed licensee event reports, licensee memoranda, unit logs, and NRC inspection reports to verify the following performance indicators for second quarter 2000 through first quarter of 2001.

- Unplanned Scrams per 7000 Critical Hours
- Scrams with Loss of Normal Heat Removal
- Unplanned Power Changes per 7000 Critical Hours
- Safety System Unavailability, High Pressure Injection System
- Safety System Unavailability, Reactor Core Isolation Cooling
- Safety System Unavailability, Residual Heat Removal System
- Safety System Functional Failures

b. Findings

No findings of significance were identified.

.4 Physical Protection Performance Indicator Verification

a. Inspection Scope

The inspector verified the data for the Physical Protection Performance Indicators (PI) pertaining to Fitness-For-Duty Personnel Reliability, Personnel Screening Program, and Protected Area Security Equipment. Specifically, a sample of plant reports related to security events, security shift activity logs, fitness-for-duty reports, and other applicable security records were reviewed for the period between October 1, 2000, and June 30, 2001.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

(Closed) LER 50-341/96005-01: Emergency Equipment Cooling Water (EECW) Makeup Tank Inoperability due to Design Issue Discovered during a Self-Assessment. In 1996, the NRC Service Water Inspection Team questioned the amount of leakage from the EECW makeup system and whether the system could perform its 30 to 100-day function given the existence of the leaks. The question caused the licensee to perform a self-assessment and conclude that the plant was in an unanalyzed condition that affected both EECW trains and a plant shutdown was conducted on March 27, 1996. This was documented in Licensee Event Report 96-005 and Deviation Event Report 96-0339. From that inspection, it was determined that each makeup tank needed an independent makeup system, which included two 5/8-inch lines providing service water (SW) to the makeup tank.

The lines, which were installed under Engineering Design Package (EDP) 28251, served as the safety-related makeup sources to the makeup tank. To install the safety-related makeup lines, the licensee had to drop the nitrogen blanket pressure from 32 to 13 psig to allow service water to discharge into the tank. However, the individual who developed the EDP did not recognize that the nitrogen blanket pressure provided adequate suction pressure to the pump to prevent water column separation and subsequent water hammers on pump restarts following a loss of power.

EECW is a standby emergency system and its piping rises through various elevations resulting in the potential for water hammers. During operation, water hammers could occur due to a momentary loss-of-power condition and the pumps stop and restart. A plant support engineer recognized this while revising Design Calculation 5760 and discovered that the documentation of engineering design basis for operation of SW/EECW cross ties could not be found and wrote CARD 99-12812 to address the missing documentation. Also, on October 4, 1999, the engineer addressed the potential water hammer condition by initiating Level 3 CARD 99-17607, "Potential for Water Hammer Due to Insufficient Makeup Tank Emergency Setpoint Pressures."

CARD 99-17607 stated that a postulated water hammer was not a part of the original design of the EECW system but was created from the SW/EECW crosstie modification installed under EDP 28251. The normal EECW pressure had to be lowered from 32 to 13 psig for sufficient minimum makeup flow. An engineer who did the modification failed to recognize that the nitrogen blanket pressure also functioned to keep the system solid during a loss-of-power event. The probable cause of this failure was that it had never been a consideration and therefore was not properly documented in the system design basis.

The interim remedy for the potential water hammer condition was to modify Abnormal Operating Procedure 20.000.18, "Dedicated Shutdown," to require that the EECW makeup tank be valved in after restarting the EECW pump. The intent of this enhancement was to reduce the rate of void collapse by limiting the rate of makeup flow from the tank. The licensee issued Document Control Revision (DCR) #00-0398, Revision 27 to Abnormal Operating Procedure 20.000.18 on May 11, 2000, to incorporate the additional operator action. In the long-term, the licensee will install 15 gpm jockey pumps to maintain system pressure. The installation is planned for 2003.

Initiative, Modification Approval Form, dated June 6, 2001, lists the initiative as, "Replace EECW Make-up System Modification." Besides resolving the water hammer condition, the form stated that the modification will, among other things, resolve actions for marginal leakage.

During leakage testing, the licensee measured EECW coolant leakage through the EECW and reactor building closed cooling water boundary Valves P440F603A(B) and P4400F601A(B). These valves isolate EECW from the nonsafety-related reactor building closed cooling water system during an accident. Leakage is required to be less than 0.0384 gpm (as determined by Design Calculation 5760) to prevent gas migration of the nitrogen blanket and gas binding of the EECW pump. An eighteen-month test was done under Event Number 0665000401 to conduct Procedure 43.401.605, "Division

1 EECW Leakage," dated April 2, 2000. Step 5.1.34.7 listed the Division 1 leakage rate as 0.0340 gpm. This was a difference of 0.0044 gpm which is within the allowed acceptance criteria but marginal. Likewise, Event Number 0666000401 required an eighteen-month EECW system test per Procedure 43.401.606, "Division 2 EECW Leakage," dated April 18, 2000. Step 5.1.34.7 listed the Division 2 leakage rate as 0.0000 gpm. The proposed jockey pumps would resolve the marginal leakage condition.

Dropping EECW makeup tank nitrogen pressure from 32 to 13 psig created operational concerns for water hammers and a reduction in the leakage margin. The likelihood for a loss of power to the running EECW pump with a design basis accident was remote. However, the licensee developed sufficient guidance to prevent a water hammer event should these conditions arise. Also, although the leak rate on Division 1 EECW was marginal, the rate remained within the acceptance criteria and was considered acceptable.

Nevertheless, the failure to recognize the adverse impact of reducing the EECW makeup tank nitrogen blanket pressure to install the service water makeup modification was a violation of 10CFR 50, Appendix B, Criterion III, "Design Control." The violation had no credible impact on plant safety. Therefore, the failure to meet Criterion III was of very low safety significance and is a violation of minor significance that was not subject to enforcement action in accordance with Section IV of the Enforcement Policy. This issue was placed into the corrective action program as CARD 99-17607.

4OA5 Management Meetings

.1 Exit Meeting Summary

The inspectors presented the inspection results to Mr. O'Connor and other members of licensee management at the conclusion of the inspection on August 3, 2001, for the biennial heat sink inspection and August 10, 2001, for the Resident, Access Authorization and 10 CFR 50.59 inspections. The licensee acknowledged the findings presented. No proprietary information was identified.

KEY POINTS OF CONTACT

Resident, Access Authorization and 10CFR 50.59 Inspections

W. O'Connor, Vice President, Nuclear Generation
D. Cobb, Director, Nuclear Production
Q. Duong, Manager, Plant Support Engineering
J. Davis, Manager, Outage
T. Haberland, Manager, Maintenance
K. Hlavaty, Manager, Operations
E. Kokosky, Manager, Radiation Protection
J. Korte, Manager, Nuclear Security
J. Moyers, Manager, Nuclear Quality Assurance
J. Thorson, Senior Line Manager, Work Control
C. Heitzenrater, Assistant Manager, System Engineering
R. Fitzsimmons, Access Control Supervisor
A. Lim, Supervisor, Plant Support Engineering Mech/Civil
B. Newkirk, Supervisor, Licensing
R. Orwig, Nuclear Security Specialist
T. Stack, Supervisor, Security Operations Support
S. Hassoun, Principal Engineer, Licensing
R. Wittschen, Principal Engineer, Licensing

Biennial Heat Sink Inspection

D. Cobb, Director Nuclear Production
K. Burke, In-Service Inspection/Performance Engineering Program
S. Berry, Supervisor, System Engineering
J. Flint, Principal Engineer, Licensing

LIST OF ITEMS OPENED AND CLOSED

Opened

50-341/01-12-01	NCV	Failure to Incorporate Test Acceptance Criteria Requirements from Applicable Design Documents
50-341/01-12-02	URI	No Oil Level for the Outboard Motor Bearing on Standby Feedwater Pump A.

Closed

50-342/01-12-01	NCV	Failure to Incorporate Test Acceptance Criteria Requirements from Applicable Design Documents
50-341/96005-01	LER	EECW make-up Tank Inoperability due to Design Issue Discovered During a Self-Assessment

LIST OF ACRONYMS USED

BCDD	Base Configuration Design Document
CARD	Condition Assessment Resolution Document
CFR	Code of Federal Regulations
DCR	Document Control Revision
DRS	Division of Reactor Safety
EDP	Engineering Design Package
EECW	Emergency Equipment Cooling Water
FOS	Functional Operating Sketch
gpm	gallons per minute
IPTE	Infrequently Performed Test or Evolution
LCO	Limiting Conditions for Operations
LER	Licensee Event Report
NRC	Nuclear Regulatory Commission
NSRG	Nuclear Safety Review Group
RHR	Residual Heat Removal
OSRO	Onsite Review Organization
psig	pounds per square inch gage
SW	Service Water
UFSAR	Updated Final Safety Analysis Report

LIST OF DOCUMENTS REVIEWED

The following documents were selected and reviewed by the inspectors to accomplish the objectives and scope of the inspection and to support any findings.

1R02 Evaluation of Changes, Tests or Experiments

EDP-31331	Addition of Flexible Hose Assemblies to RWCU Pump G3303C001A Cooling Water Lines	Rev 0
ERE 31041	Replace Torus-to-Drywell Vacuum Breaker Test Actuator Solenoid Valves	Rev A
ERE 31475	Use Spare RTD for RRMG Set B Drive Motor Winding Phase B	Rev 0
ERE 31552	Equivalent Replacement for EDG Jacket Coolant Standby Pump	Rev A
01-082-UFS	UFSAR: Existing Instruments in EDG System Omitted from Figures and Associated Drawings	7/16/2001
EDP-30815	Replace the EDG Starting Air Piping from the Starting Air Compressors up to and including the Receiver Tank Inlet Check Valves	Rev 0
IPTE 01-03	New Procedure to Determine the Maximum Power Level at which the High Pressure Turbine Valves Surveillance can be Performed with Acceptable Limits	Rev 0
SOE 00-03	Design Verification Testing During Reactor Pressure Vessel System Leakage Testing - Reactor Recirculation Speed Control System Digital Control System Replacement (EDP 27412)	Rev 0
TSR-28857	Equivalent Part Evaluation and Replacement of Thermal Overload Heaters and Fuses for Replacement Motor for Valve N3016F606	3/08/2000
23.144	Torus Water Management System	Rev 43
MES31	Diagnostic, Special and Infrequently Performed Tests, or Evolutions	Rev 7
MGA01 Appendix B	Abbreviation and Acronym List	Rev 2
MLS07	Preliminary Evaluations and 10 CFR 50.59 Safety Evaluations	Rev 18
MLS14	Changes, Tests and Experiments	Rev 0

NRC-99-0037 Enclosure 2	Fermi 2 Safety Evaluation Summary Report Changes Through 5/23/2000	4/26/1999
NRC-00-0068 Enclosure 2	Fermi 2 Safety Evaluation Summary Report Changes Through 10/29/1998	11/15/2000
SE 00-0003	Revise the Maximum Allowable Closure Stroke Time for RWCU Supply Primary Containment Isolation Valves	Rev 0
SE 00-0010	Revise Loss-of-Power Relay Setpoints	Rev 0
SE 00-0016	Condensate Storage Tank Low Level Alarm for RCIC	Rev 0
SE 00-0019	Adding Statement to B 3.4.10 to Clarify Basis for Requirements of SR 3.4.10.4. Includes Partial Power Fuel Thermal Limit Analysis Assumptions in Addition to Vessel Nozzle and Recirc System Stress Considerations	Rev 0
SE 00-0021	UFSAR Table 7.6-2 General Instrumentation Information Is Revised for Drywell Leak Detection System	Rev 0
SE 00-0024	Retirement of the Continuous Monitoring Function of CRD Panel H21P450	Rev 0
SE 01-0014	MGA11, Which Is a Procedure Described in the UFSAR, Will Be Changed to Reflect the Revised Terminology of the New 10CFR 50.59 Rule, and to Reflect Revisions to Fermi Senior Management Titles.	Rev 0
CARD 01-0027	DC-4968, Volume I Revision CARD 00-10208 Action: Update DC-4968, Volume I to Accept As-is E1150F016A(B) and T2300F409 (410) Stroke-times Which Differ from UFSAR Supporting Analysis (10 CFR 50.59 Evaluation)	Rev 0
CARD 01-0098	Update EDG Vendor Manual to Show As-Built Configuration for Lube Oil Green Bands on the Sight Glasses	Rev 0
CARD 01-0119	Revise Setpoint of Switches G41N185A/B	Rev 0
CARD 01-0131	Correcting the Normal Position of Multiple TWMS Valves from “Open” to “Closed”	Rev 0
CARD 01-0143	Use of Spare RTD for Reactor Recirculation Motor Generator (RRMG) Set Drive Motor Winding Phase B	Rev 0

CARD 01-0147	HPCI Valve E4100F027 and RCIC Valve E5100F024 Are Incorrectly Identified as Notched Globe Valves	Rev 0
CARD 01-0187	Replace Drywell to Torus Vacuum Breaker Test Actuator Solenoid Valves	Rev 0
CARD 01-0193	Equivalent Replacement For EDG Jacket Coolant Standby Pump	Rev 0
CARD 01-0224	Identify Omitted Instruments on BCDD's for the Emergency Diesel Generators	Rev 0
TSR-30772	This TSR-ABN Performs the Corrective Actions Associated with the Torus Water Management System Valve Position Discrepancies Noted Within CARD 99-15780	6/12/2001
TSR-30788	HPCI System Valve E4100F027 and RCIC System Valve E5100F024 Are Incorrectly Identified As Globe Valves	6/26/2001
TSR-31156	Revise Setpoint of Switches G41N185A and B Reference CARD 00-14254	6/19/2001
TSR-31561	Identify Omitted Instruments on BCDDs for the Emergency Diesel Generators	7/13/2001
CARD 99-15780	Valve Position Indication on 6M721-5713 Marked Incorrectly According to SOP 23.144	7/04/1999
CARD 99-16701	Corrective Actions for CARD 97-11035 (Concerns w/SE 97-0015) Do Not Appear to Address OSRO/NSRG Activities	9/20/1999
CARD 00-10016	Nitrogen Supply Lines for Drywell-to-Suppression Chamber Vacuum Breaker Test Actuators Not Currently Qualified for Pressure Boundary	1/26/2000
CARD 00-17689	Implementation of New 10 CFR 50.59 Rule	7/06/2000
CARD 01-12479	Revise Procedure 23.621	4/27/2001
CARD 01-13133	Pump R3001C028, Su2 Was Replaced by Work Request 000Z003230 Without an Equivalent Replacement Evaluation	5/18/2001
CARD 01-13150	Found RRMG Set B Drive Motor Wind Phase B Spare RTD-5 Terminated Vice RTD-2	6/14/2001

CARD 01-13151	Discrepancies Between SOP 23.628 and CECO Information Regarding RRMG Set Drive Motor Winding Temperatures	6/14/2001
CARD 01-13152	Discrepancies Between CECO and Drawings Regarding RRMG Set Drive Motor Winding RTDs	6/14/2001
CARD 01-13775	Integrated Work Management Guidelines (IWMG-14), Revision 0 Transitioning Between On-line and Off-line Reviews Need Improvement	3/29/2001
CARD 01-13942	Audit Findings, 10 CFR 50.59 Reviews for Scaffolds Currently Installed in the Plant and Procedure MMA08 Upgrade to Implement the New Rule	4/18/2001
CARD 01-14770	Operation Evolution Orders Potential Deficiency	5/17/2001
CARD 01-15476	EDG Reliability Review Instruments Not Identified on the FOS Drawing	6/07/2001
CARD 01-13244[1]	EDP-30815: Inadequate Preliminary Evaluation	8/08/2001
CARD 01-16502[1]	MES31 Definition of Special Test Needs Revising to Be Consistent with 10 CFR 50.59	8/08/2001
CARD01-16503[1]	Concern with PE for IPTE 01-03	8/09/2001
CARD 01-16504[1]	Observation of Detail in 10 CFR 50.59 Documentation	8/10/2001

[1] CARDs issued during the 10 CFR 50.59 inspection

1R04 Equipment Alignments

CARD 01-15906	EDG 14 Switchgear Room, NCX Relay Failed to Seat-in During Surveillance 42.307.02
Procedure 42.307.02	Logic System Functional Test of Division 2 EDG ECCS Emergency Start Circuits an Auto Trip/Bypass Circuits.

1R05 Fire Protection

UFSAR Section 9A.4.1.7	Second Floor, Zone 6, El. 613ft 6in
Drawing 6A721-2405	Fire Protection Evaluation Reactor and Auxiliary Buildings Second Floor - Plan - El. 613ft 6in
	Cable Tray Load Spread Sheets

UFSAR Section 9A.4.1.9	Second Floor, Zone 8, El. 659ft 6in	
Drawing 6A721-2408	Fire Protection Evaluation Reactor and Auxiliary Buildings Fourth Floor Plan El. 659ft 6in	
UFSAR Section 9A.4.1.4	High Pressure Coolant Injection Pump and Turbine and Control Rod Drive Room, Zone 3, El. 540ft 0in	
Drawing 6A721-2402	Fire Protection Evaluation Reactor and Auxiliary Buildings Basement - Plan - El. 562ft 0in	
Drawing 6A721-2401	Fire Protection Evaluation Reactor Building Sub Basement - Plan - El. 540ft 0in	
Drawing 6A721-2400	Fire Protection Evaluation Plot Plan	
Maintenance Conduct Manual MMA10	Plant Housekeeping	Rev 7
Fire Protection Procedure 28.508.05	Monthly Fire Protection Inspection	Rev 8
Engineering Support Conduct Manual MES 35	Fire Protection	Rev 2
System Operating Procedure	Fire Suppression System	Rev 31
UFSAR Section 9A.4.1.9	Fourth Floor, Zone 8, El. 659ft 6in	
Drawing 6A721-2408	Fire Protection Evaluation, Reactor and Auxiliary Buildings, Fourth Floor Plan	Rev Q

1R07 Heat Sink Performance

47.205.01	Residual Heat Removal Division 1 (North) Heat Exchanger Performance Test Accomplished September 8, 1998	Rev 6
47.205.01	Residual Heat Removal Division 1 (North) Heat Exchanger Performance Test Accomplished April 18, 2000	Rev 6
47.205.02	Residual Heat Removal Division 2 (South) Heat Exchanger Performance Test Accomplished April 2, 2000	Rev 6
CARD 98-10344	Errors Made During Revision K Update On Drawing I-2791-01	1/27/98

CARD 98-16054	Corrosion Nodules Lodged in HTX Tubes	8/25/98
CARD 00-00084	Jacket Cooling Heat Exchanger Leak	2/4/00
CARD 00-15292	Black Iron Oxide Found To Be Plugging Tubes In Division 1 CC Room Cooler	4/7/00
CARD 00-14396	Surface Indications Found During ISI NDE Magnetic Particle Exam	5/31/00
CARD 01-13239[2]	NRC Identified Incorrect LMTD Correction Factor Used in RHR Heat Exchanger Test Analysis	7/16/01
CARD 01-13240[2]	NRC Identified RHR Heat Exchanger Test Acceptance Criteria Incorrect	8/2/01
CARD 01-13241[2]	NRC Identified RHR Heat Exchanger Design Fouling Less than Fouling Allowed in the RHR Heat Exchanger Test	8/2/01
CARD 01-14194	EDG Reliability Review - Trending of EDG Information	6/12/01
CARD 01-14196	EDG Reliability Review - Operating Logs for Critical EDG Parameters	6/12/01
CARD 01-18351[2]	Sample and Analyze Deposits in EDG Heat Exchanger	8/2/01
CARD 99-12728	Preform an End Bell Inspection and Cleaning of All Five Safety Related Pumps in the Division 2 RHR Reservoir	4/14/99
70756-B	Shell Details and Assembly Residual Heat Removal Exchanger	2
70756-G	Supports and Lifting Lugs of Residual Heat Removal Exchanger	1
70756-H	Cross-Section (Assembly) of Residual Heat Removal Exchanger	1
6M721N-2052	P and ID RHR Service Water System Division I RHR Complex	Rev AC
6M721-2084	Diagram Residual Heat Removal (RHR) Division I	Rev BA
6M721-5734	Emergency Diesel Generator System Functional Operating Sketch	Rev AC
VMS21-2.0	Fromson Heat Transfer, LTD, Installation, Operation and Maintenance Instructions for RHR Heat Exchangers	4/9/83

VMS21-2.1	Fromson Heat Transfer LTD Type AEU Heat Exchanger	4/9/83
3035A01320	Calibration Record Of Data Acquisition System with 32 PRTs	9/18/00
Work Request W836961108	Internal Inspection of Service Water Cooling System IAW Generic Letter 89-13 and for Microbiologically Influenced Corrosion per DER 94-0756	4/7/98
PGT-FNPP-05	Evaluation of the Emergency Equipment Cooling Water Heat Exchanger Design Limiting Conditions	Rev 0
CARD 01-13045	DC-5931 Assumptions Potentially Impacting Emergency Equipment Cooling Water	5/18/01
CARD 01-11949	Division 2 Emergency Equipment Cooling Water Heat Exchanger Performance Is at the Operability Limit for Projected Heat Transfer Capability	6/28/01
PTP 47.207.02	Emergency Equipment Cooling Water Division 2 Heat Exchanger Performance Test	Rev 30
WR TG13000613	Perform 47.207.02 Heat Exchanger Performance Test (Emergency Equipment Cooling Water Division 2)	6/13/00
Vendor Manual VMB9-37	Alfa Laval MX25-BFD Plate Heat Exchanger Instruction Manual	Rev 0
Design Calc DC-5806	Emergency Equipment Cooling Water Design Basis Requirements Calculation	Rev 0
CARD 01-13222	Current "Calibrated" Division 2 Emergency Equipment Cooling Water Hydraulic Model May Not Accurately Predict Division 2 Emergency Equipment Cooling Water Flows	7/23/01
PTP 24.208.02	Division 1 Emergency Equipment Service Water Pump and Valve Operability Test	Rev 40
PTP 24.207.08	Division 1 Emergency Equipment Cooling Water Pump and Valve Operability Test	Rev 52
PEP Evaluation 00-007	Develop of Thermal Performance Model for Emergency Equipment Cooling Water Plate Frame Heat Exchangers Using Proto-HX Software	5/18/00

[2] CARDs issued during the biennial heat sink inspection.

1R11 Licensed Operator Requalification

SS-OP-202-0131	Loss of 345 KV Offsite Power	Simulator Scenario 4
AOP 20.000.01	Acts of Nature - Flooding	
AOP 20.300.345KV	Loss of 345KV	
AOP 20.000.21	Reactor Scram	
EOP 29.100.01	RPV Control	

1R12 Maintenance Rule Implementation

System Health Report	Reactor Controls, C1100
SOP 23.623	Reactor Manual Control System
GOP 22.000.03	Power Operation 25 Percent to 100 Percent to 25 Percent
GOP 22.000.04	Plant Shutdown From 25 Percent Power
Performance Eval Procedure 27.106.03	Control Rod Drive Insert and Withdrawal Stall Flow Measurement
Surv Procedure 24.106.01	Operable Control Rod Check
Work Request 000Z003269	Replace RWM PC, Keyboard and Monitor per ERE-31198
Work Request 000Z000741	Technical Specification Rod Position Indication Problem
Work Request 000Z011953	Faulty Light on Rod Select Switch
Work Request 000Z010275	Replace Rod Select Switches for Rods 54-31and 42-15 on H11P603
CARD 98-12429	Wrong Control Rod Position Displayed on 3DM, PC Alarm Typewriter, Full Core Display
CARD 01-11402	Various Rods Could Not be Selected
CARD 00-14838	Precaution/Limitation 3.19 of 23.623 Needs Clarification

CARD 01-14303	Control Rod Insertion Without Movement of the Rod Movement Control Switch
CARD 01-15401	Revise 22.000.03 to Include Control Rod Movement Requirement Precaution/Note (ops procedure)
CARD 01-15402	Revise 22.000.04 to Include Control Rod Movement Requirement Precaution/Note (ops Procedure)
CARD 01-15403	Revise 27.06.03 to Include Control Rod Movement Requirement Precaution/Note (ops procedure)
CARD 01-15404	Revise 24.06.01 to Include Control Rod Movement Requirement Precaution/Note (ops procedure)
CARD 97-11760	CRD has High Temperature Alarm In
CARD 00-00371	Sticky Limit Switch on C11-F127 (and associated Work Request 000Z001051
CARD 00-01020	Through Seat Leakage (and associated Work Request 000Z001513)
CARD 00-01131	C11-F113 Seat Leakage (and associated Work Request 000Z0011872)
CARD 00-01152	Packing Leak (and associated Work Request 000Z001896)
CARD 00-01155	C11-F113 Seat Leakage (and associated Work Requests 000Z0011871 and 000Z001881)
CARD 00-11323	High Temperature on CRD (and associated Work Request 000Z000851
CARD 00-17176	HCU 18-43 Scram Valve Blue Light did not Function During Scram Time Test (and associated Work Request 000Z001855)
CARD 00-17279	Control Rod 26-55 is Difficult to Withdraw (and associated Work Request 000Z002012)
CARD 00-17391	Rod Drift Alarm on Rod 46-07 (and associated Work Request 000Z002327)
CARD 00-21037	Replace Flange O-Rings CRDM 18-47 (and associated Work Request 000Z003926)
CARD 01-11402	Various Rods Could not be Selected (and associated Work Requests 000Z010275 and 000Z011463)

CARD 01-15309	West CRD Pump Motor Oil Drain Plug (and associated Work Request 000Z011701)	
CARD 98-02193	Valve Operability Extremely Difficult - C11F113 (and associated Work Request 000Z984083)	
CARD 98-18131	Obtain Actual Electrical Load on Cable 201844-OC Feeding Panel H11P615 (and associated Work Requests 000Z982952 and 000Z983610)	
CARD 98-18878	Rework C11F002B Valve Internals (and associated Work Request 000Z984009)	
CARD 98-19065	Seat/Disc Erosion and Setpoint Change for C11F002A and C11F002B (and associated Work Request 000Z991413)	
CARD 99-11982	CRD 26-39 High Temperature Alarm (and associated Work Request 000Z991445).	
CARD 99-18177	High Temperature Alarm Coming in and Clearing Often on CRD 26-31 (and associated Work Requests 000Z970249 an 000Z993941)	

<u>1R13</u> <u>Maintenance Risk Assessments and Emergent Work Evaluation</u>

CARD 01-15906	EDG 14 Switchgear Room NCX Relay Failed to Seat-in During Surveillance 42.307.02	
Surv Procedure 42.307.02	Logic System Functional Test Division 2 EDG ECCS Emergency Start Circuits and Auto Trip/Bypass Circuits	
Troubleshooting Form	EDG 14 4B Start Relay “Seal-In”	
Operations Dept Expectation ODE-2	Operations Conduct, Operations Evolution Order	Attach. 1

<u>1R15</u> <u>Operability Evaluations</u> <u>.01</u>

CARD 01-17269	Standby Feedwater Inboard Bearing Lube Oil Supply Line has Excessive Oil Flow	07/27/01
CARD 01-17025	Standby Feedwater Bearing Housing Lube Oil Supply Line is Leaking During System Standby	07/26/01
Engineering Functional Analysis	For CARD 01-17025	08/02/01

Vendor Manual VMS 23-1	Standby Feedwater Pumps, Motor and Lubricating Oil System	04/15/93
SOP 23.107.01	Standby Feedwater System	Revision 28
Ops Procedure 27.000.05	Operator Rounds	Revision 18
System Training Manual ST-OP 315-0018-001	Standby Feedwater N2103	Revision 7
TRM 3.7.7	Alternative Shutdown Auxiliaries	Appendix R

<u>1R15</u> <u>Operability Evaluations</u> <u>.02</u>

Engineering Functional Analysis	For CARD 00-00694, Leak - P4400F129A
Engineering Functional Analysis	For CARD 01-13205, Inadvertent Loading on CTG 11-1 During Station Blackout and Dedicated Shutdown
Engineering Functional Analysis	For CARD 01-17276, EDG Broken Studs on Vertical Drive Compartment
LCO 01-0269	CTG 11 Unit 1 Operability Determination Required for Unresolved Loading Issue During Station Blackout
DER 88-0679	Broken Stud Weld on EDG 12 R30015002 Vertical Drive Inspection Cover
Work Request 000Z000616	Leak-Off Nipple is Leaking Approximately 4 dpm
Work Request 000Z012006	Division 1 EECW Leakage MES 27 Request per TMTE 99-0252
TS 5.5.6	Inservice Testing and Inspection Program
Surv Procedure 43.401.605	Division 1 EECW System Leakage Test

<u>3PP</u> <u>Physical Protection</u>

CARD 00-25205	Security Controlled Operator Round Keys Removed from Protected Area

Nuclear Security Administrative Procedure	SEP-SE1-01(Testing and Maintenance)	Rev. 17
CARD 01-14274	Adverse Trend in Security Equipment Performance	
MGA16	General Administration Conduct Manual (Behavioral Observation)	Rev 0
MGA10	General Administration Conduct Manual (Fitness for Duty)	
LP-GN-501-0001	Nuclear Training Lesson Plan (Behavior Observation)	
NQA Audit Report 00-010	Security, Safeguards Information, Access Authorization and Personnel Access Data System	7/21/00
NQA Audit Report 00-0113	Fitness-For-Duty Program	11/14/00
NSSE-00-0147	Vehicle Search Officer Duties	12/7/00
NSSE-01-0094	Truck Lock-Vehicle Search Duties	7/19/01
NSSE-00-0152	Warehouse B Material Transfer	12/20/00
Nuclear Security Compliance Evaluation Report 00-0069	SAS Access Control Officer	9/11/00
Nuclear Security Compliance Evaluation Report 00-0068	Personnel Search Officer	6/28/00
Semi-Annual FFD Report	January 1, 2000 - June 30, 2000	8/29/00
Semi-Annual FFD Report	July 1, 2000 - December 31, 2000	2/23/01
MGA09	General Administration Conduct Manual (Access Control)	Rev 13

4A02 Performance Indicator Verification

24.000.02	Shiftly, Daily and Weekly Required Surveillances	Rev 99, Attach 1

TMIS-01-0008	Nuclear Generation Memoranda	01/08/01
TMIS-01-0069	Nuclear Generation Memoranda	04/16/01
TMIS-00-0144	Nuclear Generation Memoranda	10/09/00
TMIS-00-0098	Nuclear Generation Memoranda	07/14/00
NPRC-01-0016	Nuclear Generation Memoranda	01/11/01
NPRC-01-0112	Nuclear Generation Memoranda	04/06/01
NPRC-00-0329	Nuclear Generation Memoranda	10/05/00
NPRC-00-0223	Nuclear Generation Memoranda	07/03/00
TMTE-00-0134	Nuclear Generation Memoranda	06/11/00
TMTE-00-0188	Nuclear Generation Memoranda	10/09/00
TMTE-01-0010	Nuclear Generation Memoranda	01/11/01
TMTE-01-0063	Nuclear Generation Memoranda	04/10/01
NANL-01-0045	Nuclear Generation Memoranda	04/11/01
NANL-01-002	Nuclear Generation Memoranda	01/08/01

4OA3 Event Follow-up

CARD 99-17607	Potential for Water Hammer Due to Insufficient Makeup Tank Emergency Setpoint Pressures	10/04/99
Modification Approval Form	Replace EECW Make-up System Modification	06/06/01
DER 96-0339	Potential Leak Path from EECW	04/16/98
EDP 28251	EECW Makeup Tank Modification/ EECW to EECW Makeup Tank	04/04/96
Design Calc 5760	EESW/EECW Hydraulic/N_2 Supply to EECW Makeup Tank	05/30/00
Procedure 43.401.606	Division 2 EECW Leakage	
Event 0665000401	Perform 43.401.605 EECW Division 1 EECW Leakage Test	04/0100.
Procedure 43.401.605	Division 1 EECW Leakage	
Event 0666000401	Perform 43.401.606 Division 2 EECW Leak Test	04/01/00

CARD 99-10937	Affect of SW Pump Degradation on EECW MU	02/22/99
Sargent & Lundy Letter	Project No. 09471-019, Contract 317721	05/11/98
TSR 29980	Upgrade EECW Make up Water Supply from EESW (possible jockey pump)	
Project Evaluation Review Committee	Meeting Minutes	06/06/00
CARD 00-15085	Division 1 EECW Temperature Control Valve Controller (P44K800A)	04/16/00
ECRs 29792-14 and 15	EECW Temperature Control Valve and Controller Replacement	10/27/99
TSR 31155	EECW Temperature Control Valve and Controller Replacement	10/27/99
AOP 20.000.18	Dedicated Shutdown	