June 5, 2001

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

SUBJECT: FERMI POWER PLANT
NRC INSPECTION REPORT 50-341/01-05(DRS)

Dear Mr. O'Connor

On May 4, 2001, the NRC completed an inspection at your Fermi Power Plant, Unit 2. The enclosed report documents the inspection findings which were discussed on May 4, 2001, with Mr. P. Fessler and other members of your staff.

The inspection was a detailed examination of design activities and records as they related to ensuring that the residual heat removal system and its required support systems were capable of performing required post-accident functions, and to verify compliance with the Commission's rules and regulations and the conditions of your license. Within these areas, the inspection consisted of observations of activities, discussions with cognizant personnel and a selective examination of procedures, design documents, and representative records.

Based on the results of the inspection, one finding of very low safety significance (Green) was identified. The finding involved a violation of NRC requirements. However, because of its very low safety significance and because it was entered into your corrective action program, the NRC is treating the issue as a Non-Cited Violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest the 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 Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, Region III, Resident Inspector and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

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).

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

/RA/

John M. Jacobson, Chief
Mechanical Engineering Branch
Division of Reactor Safety

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

Enclosure: Inspection Report 50-341/01-05(DRS)

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
Emergency Management Division
MI Department of State Police

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-341
License No:	NPF-43
Report No:	50-341/01-04(DRS)
Licensee:	Detroit Edison Company
Facility:	Fermi Power Plant, Unit 2
Location:	6400 N. Dixie Highway Newport, MI 48166
Inspection Dates:	April 16 through May 4, 2001
Inspectors:	P. Lougheed, Lead Inspector J. Gavula, Reactor Inspector G. O'Dwyer, Reactor Inspector W. Scott, Reactor Inspector S. Sheldon, Reactor Inspector R. Quirk, Contractor
Approved by:	John M. Jacobson, Chief Mechanical Engineering Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000341-01-05(DRS), on 04/16-05/04/01, Detroit Edison Company, Fermi Power Plant, Unit 2. Safety System Design and Performance Capability.

The inspection was routine baseline inspection of the design and performance capability of the residual heat removal system. It was conducted by regional engineering specialists. One Green finding was identified during the inspection, which involved two examples of 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." 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**. Findings for which the Significance Determination Process does not apply are indicated by "No Color" or by the severity level of the applicable violations.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

Green. The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," with two examples. The first example involved failure to assure that the test for proper function of residual heat removal keep fill check valves in the residual heat removal system could detect a failure of the valves. The second example involved failure to include a design function for the residual heat removal pump discharge check valves in the inservice testing program.

The finding was of very low safety significance because the licensee was able to show that current leakage past both groups of check valves was sufficiently small to preclude adverse effects on the residual heat removal system (1R21).

B. Licensee Identified Findings

No findings of significance were identified.

Report Details

Summary of Plant Status: The unit was at or near 100 percent power for most of the inspection. One down power occurred for an unrelated issue.

1. REACTOR SAFETY

Cornerstones: Mitigating Systems and Barrier Integrity

1R21 Safety System Design and Performance Capability (71111.21)

The residual heat removal (RHR) system was selected for review during this safety system design and performance capability inspection at the Fermi Power Plant, Unit 2. The purpose of the inspection was to assess whether the design bases had been correctly implemented and to ensure that the system could be relied upon to meet functional requirements. The inspection was performed in accordance with the Nuclear Regulatory Commission's (NRC) regulatory oversight process, which uses a risk-informed approach for selecting the risk significant areas and attributes to be inspected.

.1 System Requirements

a. Inspection Scope

The inspectors reviewed the updated final safety analysis report, technical specifications, and available design basis information to determine the performance requirements of the residual heat removal system. The reviewed system attributes included process medium, energy sources, control systems, operator actions and heat removal. The rationale for reviewing each of the attributes was:

Process Medium: This attribute needed to be reviewed to ensure that the residual heat removal system would supply the required amount of water to achieve its design basis function of cooling the core following a transient.

Energy Sources: This attribute needed to be reviewed to ensure that the residual heat removal system would start when called upon, and, that appropriate valves would have sufficient power to change state when so required.

Controls: This attribute required review to ensure that the automatic controls for starting the system were properly established. Additionally, review of alarms and indicators was necessary to ensure that operator actions would be accomplished in accordance with the design.

Operations: This attribute was reviewed because all but one of the functions of the residual heat removal system were manually initiated. Therefore operator actions played an important role in the ability of the residual heat removal system to achieve its safety related functions.

Heat Removal: This attribute required review because a function of the residual heat removal system was to remove heat from the suppression pool and core following an event. This required operation of a heat exchanger and a support system (residual heat removal service water)

b. Findings

No findings of significance were identified.

.2 System Condition and Capability

a. Inspection Scope

The inspectors reviewed information to verify that the actual system condition and tested capability was consistent with the identified design bases. Specifically, the inspectors reviewed the installed configuration, the detailed design and the system testing, as described below.

Installed Configuration: The inspectors confirmed that the installed configuration of the residual heat removal system met the design basis by performing system walkdowns. A limited walkdown of the residual heat removal service water system was also performed. The walkdowns focused on the installation and configuration of piping, components, and instruments; the placement of protective barriers and systems; the susceptibility to flooding, fire, or other environmental concerns; physical separation; provisions for seismic and other pressure transient concerns; and the conformance of the currently installed configuration of the systems with the design and licensing bases.

Design: The inspectors reviewed the mechanical, electrical and instrumentation design of the residual heat removal system to verify that the system and subsystems would function as required under accident conditions. The review included a review of the design basis, design changes, design assumptions, calculations, boundary conditions, and models as well as a review of selected modification packages. Instrumentation was reviewed to verify appropriateness of applications and set-points based on the required equipment function. Additionally, the inspectors performed limited analyses in several areas to verify the appropriateness of the design values.

Testing: The inspectors reviewed records of selected periodic testing and calibration procedures and results to verify that the design requirements of calculations, drawings, and procedures were incorporated in the system and were adequately demonstrated by test results. Test results were also reviewed to ensure automatic initiations occurred within required times and that testing was consistent with design basis information. Pre-operational test data was also reviewed to confirm initial design parameters that could not be tested under normal operations.

b. Findings

(1) Inadequate Test Control

The inspectors identified a Green finding relating to the licensee's testing of check valves in the RHR system. The finding had two examples of a Non-Cited Violation for Inadequate Test Control. Failure of the valves could result in insufficient RHR water being provided to the reactor during design basis events. Each example is discussed separately below.

Inadequate Test to Verify Check Valve Closure

The design function of RHR keep fill check valves E1100-F089, -F090, -F184, and -F185 was to close in order to prevent system flow from being diverted into the condensate storage and transfer (CST) system. Surveillance procedure 24.204.01, step 5.1.21, verified that these valves were functioning properly through observation of a relief valve in the CST system. The rationale behind the procedure step was that if the check valves had properly seated, then the relief valve would not lift. If the check valve were not functioning properly, then the higher pressure RHR water would enter the CST system and lift the relief valve. The test was inadequate in that the procedure did not require isolation of the CST system when performing the test in order to ensure that the pressure would increase if the check valves were not functioning. The CST system had normally operating loads that could have provided an unrestricted flow path from the residual heat removal system if the check valves were not functioning properly. Because these flow paths were not isolated during the testing, any check valve leakage could pass into the condensate storage and transfer system without resulting in the relief valve lifting, and thus remain undetected.

The inspectors considered this issue to have a credible impact on safety because the failure of these check valves might have prevented the RHR system from meeting its design flow and pressure requirements. The licensee acknowledged the test deficiency, wrote a corrective action resolution document (CARD) and performed an operability evaluation. The licensee determined that check valve leakage severe enough to impede the RHR post-accident function would have been detected through alternate means. The inspectors agreed with the licensee's assessment that the alternate means would have detected severe valve leakage. The inspectors determined that the issue involved the mitigating system cornerstone and evaluated the issue against the Phase I significance determination process (SDP) criteria. The issue screened out as Green.

10 CFR Part 50, Appendix B, Criterion XI requires, in part, that testing demonstrate that components will perform satisfactorily in service and that the testing be accomplished in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Contrary to the above, as of May 4, 2001, the NRC identified that surveillance test 24.204.01 did not demonstrate that RHR keepfill check valves would perform satisfactorily in service because the test did not establish appropriate acceptance limits. This issue is a violation of 10 CFR Part 50, Appendix B, Criterion XI. In accordance with Section VI.A.1 of the NRC Enforcement Policy, this violation is being treated as an example of a Non-Cited Violation (NCV 50-341-01-05-01a). It was entered into the licensee's corrective action program as CARD 01-13989.

Inadequate Design Safety Function Specified for Pump Discharge Check Valves

The licensee had identified only one design function for RHR pump discharge check valves E1100-F031A, -F031B, -F031C, and -F031D: to close in order to prevent flow diversion through a non-running pump. However, an additional design function of these check valves was to prevent excessive leakage that could result in void formation in the upper elevations of the system and make the system susceptible to significant pressure transients. The first design function was properly specified and verified during surveillance procedure 24.204.01. However, the second design function was not in the licensee's inservice testing program and did not have an inservice test acceptance criteria in surveillance procedure 24.204.01. The leakage design function and acceptance criteria had previously been included in surveillance procedure 24.204.01, but were removed in 1995.

The inspectors considered this issue to have a credible impact on safety in that excessive leakage through the pump discharge check valves would drain portions of the RHR system, leaving it susceptible to significant pressure transients. Although the licensee had scoped the size of the pressure transients, they had not evaluated the consequences on the system. However, given the postulated size of the transient, the inspectors deemed that the safety function of the RHR system could be impacted. The licensee acknowledged the design function of the valves had not been properly incorporated into the inservice test program and wrote a CARD. The licensee was able to show that the check valve leakage was minimal at the time of the inspection because the system engineer had continued to check for leakage during performance of surveillance procedure 24.204.01, even though the acceptance criteria had been removed from the inservice testing program. The inspectors determined that the issue involved the mitigating system cornerstone and evaluated the issue against the Phase I significance determination process (SDP) criteria. The issue screened out as Green.

10 CFR Part 50, Appendix B, Criterion XI requires, in part, that testing demonstrate that components will perform satisfactorily in service and that the testing be accomplished in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Contrary to the above, as of May 4, 2001, the NRC identified that surveillance test 24.204.01 did not demonstrate that RHR pump discharge check valves would perform satisfactorily in service because the test did not establish appropriate acceptance limits. This issue is a violation of 10 CFR Part 50, Appendix B, Criterion XI. In accordance with Section VI.A.1 of the NRC Enforcement Policy, this violation is being treated as an example of a Non-Cited Violation (NCV 50-341-01-05-01b). It was entered into the licensee's corrective action program as CARD 01-13948.

(2) Risk Associated with High Drywell Pressure Due to Loss of Offsite Power

The inspectors identified an unresolved issue (URI) related to the increased use of the torus (suppression pool) cooling mode during normal operation. The licensee had been entering the torus cooling mode more frequently than usual due to problems with the high pressure coolant injection system. The NRC had previously identified a potential for a hydraulic transient when licensees were in the torus cooling mode.

The NRC's Information Notice No. 87-10 alerted licensees to the potential for having a hydraulic transient in the RHR system if a design basis loss of coolant accident coincident with a loss of offsite power occurred while the plant was using torus cooling. At the time of the information notice, many plants were using torus cooling to mitigate the impact of leaking safety relief valves. The licensee's conservative evaluation of the potential hydraulic transient, in calculation DC-5596, indicated that pipe segment loads as high as 180,000 pounds could be generated, potentially disabling the system. The licensee took several steps to resolve the concern including modifying the start logic for the RHR pumps such that the pumps would not automatically load onto the emergency diesel generators after a loss of offsite power. The licensee also repaired all leaking safety relief valves and committed not to operate in the torus cooling mode more than 5.5% of the time (approximately 482 hours a year). This percentage was based on the maximum time spent in torus cooling at the time the commitment was made. The licensee performed a probabilistic analysis which determined that the likelihood of a design basis loss of coolant accident coincident with a loss of offsite power was extremely remote. When combined with the limited time spent in this mode, the licensee determined that the impact on core damage frequency was minimal.

During the inspection, the inspectors ascertained that a comparable hydraulic transient could occur during a loss of offsite power while in the torus cooling mode if drywell cooling were to be lost (i.e., without a concurrent loss of coolant accident). In this scenario, the increasing drywell temperature would cause the drywell pressure to exceed the set point of 1.68 pounds per square inch in less than two minutes. The drywell pressure signal would initiate all four RHR pumps, causing a potentially disabling hydraulic transient that would significantly affect the containment heat removal capability. The licensee evaluated the risk associated with this new scenario and determined it to be of very low risk, due to the drywell coolers automatically loading on to the diesels after a diesel start. However, the risk was higher than previously considered.

Additionally, the inspectors determined that the system could be at risk due to the operators manually starting the system. Upon a loss of offsite power, the main steam valves would close, and the safety relief valves would discharge into the torus, raising the pool temperature. The emergency operating procedures for containment call for the operators to use all available RHR pumps in the torus cooling mode to keep the torus temperature below 95 degrees. Although the operating procedure, 23.205, contained a note to fill and vent the RHR system if the system was running and then lost, the note did not appear at the start of section for emergency use of torus cooling. Furthermore, the procedure divided operation of the torus cooling mode into a normal and an emergency mode. Therefore, the inspectors deemed it a plausible scenario for the RHR system to be running in torus cooling mode during normal operation, for a loss of offsite power to occur, and for the shift manager to order start of the torus cooling mode in the emergency mode without filling and venting the system.

The inspectors considered this issue to have a credible impact on safety because the licensee had not considered the increased risk impact of running the RHR system in torus cooling mode for events other than leaking safety relief valves. For example, at the time of the inspection, the licensee was using the torus cooling mode because of increased surveillances on the high pressure coolant injection system. Additionally, the licensee had not evaluated the risk associated with either the RHR system automatically starting on high drywell pressure or being manually started following a loss of offsite power. The licensee wrote CARD 01-14730 and was evaluating the increased risk significance of the issue at the end of the inspection. The issue is being left as an unresolved item (URI 50-341-01-05-02) pending NRC review of the overall risk associated with this issue. In addition, the Office of Nuclear Reactor Regulation (NRR) will be consulted on the generic aspects of this concern.

.3 Components

a. Inspection Scope

The inspectors examined the residual heat removal system to ensure that component level attributes were satisfied. The attributes selected for review were: equipment and environmental qualification, equipment protection and operating experience.

Equipment and Environmental Qualification: To confirm this attribute, the inspectors reviewed calculations and equipment qualification documents to ensure that components in the residual heat removal system would perform their function under the temperatures that would be expected.

Equipment Protection: The inspectors reviewed calculations and other documents, performed walkdowns and interviewed personnel to ensure that components located in the residual heat removal rooms would perform their function following seismic and high and moderate energy line break events.

Operating Experience: The inspectors reviewed condition assessment resolution documents, licensee responses to industry events and NRC generic communications, and other documents to confirm that the licensee adequately evaluated industry information regarding residual heat removal problems.

b. Findings

No findings of significance were identified.

.4 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed a sample of residual heat removal system problems identified in the licensee's corrective action program. The inspectors also evaluated the licensee's responses to issues identified during the inspection and condition assessment resolution documents generated as a result. This review was performed to evaluate the adequacy and effectiveness of the identification and correction of residual heat removal system problems. Inspection Procedure 71152, “Identification and Resolution of Problems,” was used as guidance for inspection in this area.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA6 Meeting

Exit Meeting

The inspector presented the inspection results to Mr. P. Fessler and other members of licensee management at the conclusion of the inspection on May 4, 2001. The licensee acknowledged the findings presented. The inspectors identified the proprietary information reviewed during the inspection and questioned the licensee as to whether proprietary information had been retained. The inspectors also discussed the potential for proprietary information to be included in the inspection report. The licensee confirmed that no proprietary information was retained at the completion of the inspection. The licensee concurred that the proposed inspection report content would not compromise any proprietary information.

KEY POINTS OF CONTACT

Licensee

M. Baxi, Electrical Engineer, Plant Support Engineering
L. Biehle, Mechanical/civil Engineer, Plant Support Engineering
Q. Duong, Electrical Lead, Plant Support Engineering
P. Fessler, Plant Manager
K. Harsley, Licensing
R. Haupt, System Engineer
K. Howard, Director, Plant Support Engineering
A. Klemptner, Instrumentation and Controls Engineer, Plant Support Engineering
R. Libra, Technical Manger
J. Moyers, Director, Nuclear Quality Assurance
K. Noetzel, Director, System Engineering
N. Peterson, Director, Licensing
M. Thrift, Operation Support Supervisor, Operations

NRC

R. Caniano, Deputy Director, Division of Reactor Safety
J. Jacobson, Chief, Mechanical Engineering Branch
J. Larizza, Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed in This Inspection

50-341/01-05-01	NCV	Test procedure did not contain appropriate acceptance limits to assure check valve function; two examples of 10 CFR Part 50, Appendix B, Criterion XI

Opened

50-341/01-05-02	URI	Risk associated with operation in torus cooling mode beyond original design assumptions

LIST OF ACRONYMS USED

ADAMS	Agency-wide Documents and Management System
CARD	Corrective Action Resolution Document
CFR	Code of Federal Regulations
CST	Condensate Storage and Transfer
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling System
LPCI	Low Pressure Coolant Injection
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PARS	Publically Available Records
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
SDP	Significance Determination Process

LIST OF DOCUMENTS REVIEWED

Calculations

Number	**Title**	**Revision or Date**
DC-0367	Design Calculations for the Residual Heat Removal (RHR) System, Volume 1	M
DC-0592	RHR System Run-out Assessment Design Comparison	A
DC-0835	System Voltage Study	D
DC-0885	Emergency Core Cooling System (ECCS) Suction Line Air Ingestion	C
DC-0919	Undervoltage Relay Setpoints	D
DC-2689	Flow Section - Sizing and Instrument Calibration Verification for Heat Exchanger Service Water Flow	0
DC-3122	Piping Stress Report X210A AND X211A	0
DC-3123	Piping Stress Report X210B AND X211B	0
DC-3222	Class 1E Equipment Qualification Review RHR System	
DC-4522	Reactor Dome Pressure Instrumentation Surveillance Procedure Validation, Volume I	F
DC-4523	Reactor Wide Range Water Level Surveillance Procedure Validation, Volume I	G
DC-4528	Reactor Narrow Range Water Level Surveillance Procedure Validation, Volume I	E
DC-4529	Drywell Pressure Surveillance Procedure Validation, Volume I	F
DC-4540	RHR Pump Discharge Pressure Surveillance Procedure Validation, Volume I	D
DC-4557	RHR Flow Surveillance Requirement, Volume I	F
DC-4588	RHR Keep Fill, Volume I	0
DC-4607	Surveillance Calibration Validation for Reactor Recirculation Pump Suction Pressure Instrumentation, Volume I	B
DC-5036	Maximum Expected Differential Pressure for Motor Operated Valves, Volumes I and XV	A & 0
DC-5038	Torque/Thrust Calculation for E1150F048 A & B	A & C

Calculations

Number	Title	Revision or Date
DC-5079	RHR & Core Spray System Technical Specification Surveillance Pump Discharge Pressure, Volume I	A
DC-5110	High Energy Line Break, Volume I	A
DC-5141	Low Pressure Coolant Injection (LPCI) and Core Spray Flows, Volume I	0
DC-5268	Motor Operated Valves Available Terminal Voltage, Attachment I, Volume I	L
DC-5272	Sizing Criteria and Basis for Fuses Used In Power Distribution System and Control Circuits	B
DC-5325	Evaluation of Reactor Water Cleanup Line Break for Power Uprate	0
DC-5331	High Energy Line Break Profiles for Equipment Qualification for Power Uprate	0
DC-5349	Alternating Current Control Cable Voltage Drop Calculation for QA1, Division-1	E
DC-5350	Alternating Current Control Cable Voltage Drop Calculation for QA1, Division II, Volume I	E
DC-5352	Direct Current Control Cable Voltage Drop Calculation for QA-1 Division II	C
DC-5426	Moderate Energy Line Break	A
DC-5455	Environmental Response Evaluation, Volume I	B
DC-5474	Hydraulic Transient Analysis for the RHR Service Water System (RHRSW), Volume I	A
DC-5573	Starting Torque at Elevated Temperature	D
DC-5589	Reactor Building Environmental Response for High Energy Line Break and Loss of Coolant Accident, Volume I	A
DC-5596	Hydraulic Transient Evaluation for RHR During Suppression Pool Cooling, Volume Ia	0
DC-5707	RHRSW Channel Instrument Error	B
NEDC 31982P	Fermi 2 SAFER/GESTR - Loss of Coolant Accident Analysis GE Nuclear Energy	July 1991
NEDC-31336P-A	General Electric Instrument Setpoint Methodology, Class 3	September 1996

Condition Assessment Resolution Documents Generated Due to the Inspection

Number	Title	Revision or Date
01-12480	Needs Analysis – NRC Safety System Design Inspection (SSDI) Concern	May 4, 2001
01-13043	Revise 6M721-5706-1 and 6M721-5706-2	May 15, 2001
01-13082	Calculation Did Not Assume Path of Least Resistance in Determining RHR Pump Runout Flow	May 4, 2001
01-13096	Apparent Errors in Instrument Accuracy Data Used in Instrument & Control Surveillance Procedure Validation Design Calculation	May 2, 2001
01-13097	Potential Inaccurate Design Input/Data Used in Calculations DC-4588 and DC-5707	May 7, 2001
01-13098	Questions on Calculation DC-5036, Volume XV, Revision A	May 4, 2001
01-13099	Procedure 20.208 Does Not Appear to Account for Instrument Uncertainties in the Flow Values Listed	May 4, 2001
01-13101	Revised Orifice Maximum Flow DC-0367	May 4, 2001
01-13169	Errors in Moderate Energy Line Break Evaluation Calculation DC-5426	May 2, 2001
01-13948	NRC Concern on Design Safety Function of Check Valves in RHR System	May 2, 2001
01-13989	Potentially Inadequate Closing Test (CT-C) for RHR Keep Fill Check Valves	April 26, 2001
01-14697	Potential Enhancement to RHRSW Monitoring for Heat Exchanger Tube Leakage Detection	May 4, 2001
01-14720	Training Material ST-OP-315-0041-001 Requires Updating	May 2, 2001
01-14721	Training Material IC-331-1001 Requires Updating	May 2, 2001
01-14727	NRC Concern – RHR Heat Exchanger Monitoring	May 4, 2001
01-14728	NRC Concern – Suppression Pool Cooling Maximum Flow	May 4, 2001
01-14730	Risk Significance of a Loss of Offsite Power with RHR in Torus Cooling Mode	May 4, 2001

Condition Assessment Resolution Documents Reviewed During the Inspection

Number	Title	Revision or Date
DER 94-0328	Inadequate Pick-up Voltage Testing of RHR Time Delay Relays	July 26, 1994
DER 97-0747	Report on Information Notice (IN) 96-55	May 7, 1997
97-08366	Degraded Jacket on Cables	December 4, 1997
97-11443	NRC Generic Letter (GL) 97-04, Assurance of Net Positive Suction Head For Emergency Core Cooling and Containment Heat Removal Pumps	October 13, 1997
98-16038	Thermal Overloads on the New Motor Control Center Positions May Not Have Been Tested Sufficiently to Satisfy Technical Specifications	July 15, 1998
98-18661	Division I RHR Heat Exchanger Flow Rate Exceeded 10,700 Gallons per Minute by Indication on Temporary Flow Meter	October 9, 1998
99-13295	Questions Regarding The Use of RHR Fuel Pool Cooling and Cleanup Assist and Shutdown Cooling Split Flowpath Modes	April 28, 1999
99-12842	Missed Redlining of Control Room Drawing I-2205-12 for Division I for EDP 28998	March 30, 1999
00-10208	Mechanical Design Calculation Review Finding Against DC-4968	July 14, 2000
00-10552	HFA Relays Fail As-Found Test	February 10, 2000
00-10863	Blown Division II RHR Logic Fuse Due to Shorting of Limit Switch Contacts	April 27, 2000
00-10990	Damping Adjustment Specification Requirement for E11N021A	August 3, 2000
00-11017	Loss of Power to E1150-F015A; Motor Control Center 72CF, Position 2C Open	January 07, 2000
00-11415	RHR A Start Delay (Pump)	April 16, 2000
00-15146	Loss of Shutdown Cooling	April 22, 2000
00-15332	Schematic for E1150-F015A is Incorrect	April 28, 2000
00-16205	Test Gauge Damaged During 24.204.06	May 4, 2000
00-16946	PM Program Deficiencies	June 22, 2000
00-17616	B31N6111B Analog Indicator Could Not be Calibrated	June 14, 2000
00-18253	4160 VAC Breaker Failed Trip Coil Acceptance Criteria Pickup Voltage	August 10, 2000

Condition Assessment Resolution Documents Reviewed During the Inspection

Number	Title	Revision or Date
00-18266	Scheduling Error of PM E414941008 During RHR Safety System Outage	August 08, 2000
00-19481	Performance Criteria Exceeded for RHR System	September 21, 2000
00-19680	Significant Degradation of Motor Actuator has Caused High Out-of-Spec Stroke Time	September 19, 2000
00-19686	E1150f024b Flow Inconsistencies in the Full Open Position	September 23, 2000
01-10678	Standby Feedwater Surveillance Testing and Preconditioning	March 12, 2001
01-12816	RHRSW Pump Injection to Reactor Pressure Vessel May Not Be Adequate for Reactor Pressure Vessel Flooding	March 13, 2001

Correspondence

Number	Title	Revision or Date
96-128	Operational Experience Disposition Memorandum - Review of Service Information Letter (SIL) 603	November 11, 1996
97-017	Independent Safety Engineering Group Review Record Review of DM 96-128 (SIL 603)	September 24, 1998
Internal Memo	J. Ramirez to K. Harsley: Risk Significance of a Loss of Offsite Power Producing a High Drywell Pressure While RHR is in Torus Cooling	May 4, 2001
Letter	GE, K. Taghavi to DECo, L Frasson: Updated UFSAR Figure 7.5-10 - Maximum Time Available for Operator Action	July 30, 1996
Letter NRC-97-0114	Detroit Edison, D Gipson, to NRC: 30-Day Response to NRC Generic Letter 97-04	November 5, 1997
Letter NRC-97-0143	Detroit Edison, D Gipson, to NRC: 90-Day Response to NRC Generic Letter 97-04	December 30, 1997
Letter NRC-98-0145	Detroit Edison, D Gipson, to NRC: 120 Day Response to Generic Letter No. 98-04	November 11, 1998
SIL 603	Limiting Conditions for Net Positive Suction Head Calculations	September 19, 1996

Design Basis Documents

Number	Title	Revision or Date
E11-00	Residual Heat Removal System	A
E11-XX	Residual Heat Removal Service Water System	A

Drawings

Number	Title	Rev or Date
6I721-2045-60	Internal-External Wiring Diagram Division II Core Spray Cab H11-P027	R
6I721-2095-14	Schematic Diagram Nuclear Steam Supply System Shut Off System Trip System A	M
6I721-2095-29	Schematic Diagram NBS ANN Inputs & Relay Tabulation Testability Modification	T
6I721-2095-30	Schematic Diagram Nuclear Boiler Process Instrument A & B Circuits Testability Modification	L
6I721-2095-37	Schematic Diagram Nuclear Boiler Process Instrument A & B Circuits - Part II Testability Modification	H
6I721-2105-11	Schematic Diagram Reactor Recirculation Pump & Motor-Generator Set Testability Modification	G
6I721-2105-12	Schematic Diagram Reactor Recirculation Pump & Motor-Generator Set Testability Modification	J
6I721-2200-01	Logic Diagram RHR System	B
6I721-2200-02	Logic Diagram RHR System	D
6I721-2200-03	Logic Diagram RHR System	D
6I721-2200-04	Logic Diagram RHR System	D
6I721-2200-05	Logic Diagram RHR System	F
6I721-2200-06	Logic Diagram RHR System	B
6I721-2200-07	Logic Diagram RHR System	B
6I721-2200-08	Logic Diagram RHR System	B
6I721-2200-09	Logic Diagram RHR System	B & C
6I721-2200-10	Logic Diagram RHR System	C
6I721-2200-11	Logic Diagram RHR System	D
6I721-2200-12	Logic Diagram RHR System	A
6I721-2200-13	Logic Diagram RHR System	0
6I721-2201-01	Schematic Diagram RHR Pump A	R

Drawings

Number	Title	Rev or Date
6I721-2201-02	Schematic Diagram RHR Pump B	N
6I721-2201-03	Schematic Diagram RHR Pump C	R
6I721-2201-05	Schematic Diagram Reactor Recirculation Extractor to RHR Outboard Valve E1150-F008	Y
6I721-2201-06	Schematic Diagram RHR Heat Exchanger B Outlet Valve E1150-F003B	K
6I721-2201-07	Schematic Diagram Suppression Pool to Pump B Valve E1150F004B	P
6I721-2201-08	Schematic Diagram Suppression Pool to Pump D Valve E1150F004D	N
6I721-2201-09	Schematic Diagram Recirculation Line to Pump A & B Valves E1150-F006A, E1150-F006B	L
6I721-2201-10	Schematic Diagram Recirculation Line to Pump D Valve E1150-F006D	K
6I721-2201-11	Schematic Diagram Loop A & B Minimum Flow Bypass Valves E1150-F007A & E1150-F007B	U
6I721-2201-12	Schematic Diagram Reactor Recirculation Extractor to RHR Valve E1150-F009	R
6I721-2201-13	Schematic Diagram Cross Tie Header Valve E1150-F010	M
6I721-2201-15	Schematic Diagram Head Spray Inboard Isolation Valve E1150-F022	Q
6I721-2201-16	Schematic Diagram Head Spray Outboard Isolation Valves E1150-F023 E1150-F049	R
6I721-2201-17	Schematic Diagram RHR Loop B Recirculation Inboard Isolation Valve E1150F015B	N
6I721-2201-18	Schematic Diagram Containment Spray Outboard Isolation Valve E1150-F016B	R
6I721-2201-19	Schematic Diagram RHR Loop B to Recirculation Outboard Isolation Valve E1150F017B	M
6I721-2201-20	Schematic Diagram Containment Spray Inboard Isolation Valves E1150-F021A & E1150-F021B	N
6I721-2201-21	Schematic Diagram Suppression Chamber Spray Bypass Isolation Valve E1150-F024B	T
6I721-2201-22	Schematic Diagram Suppression Chamber Spray Bypass Isolation Valve E1150-F024B	T

Drawings

Number	Title	Rev or Date
6I721-2201-23	Schematic Diagram RHR Warmup Line Isolation E1150-F026B	M
6I721-2201-24	Schematic Diagram Suppression Chamber Spray Inboard Isolation Valves E1150-F027A & E1150-F027B	O
6I721-2201-25	Schematic Diagram Suppression Chamber Spray Outboard Isolation Valve E1150-F028B	T
6I721-2201-25	Schematic Diagram Suppression Chamber Spray Outboard Isolation Valve E1150-F028B	T
6I721-2201-27	Schematic Diagram RHR Heat Exchanger Inlet Valve E1150F047B	I
6I721-2201-28	Schematic Diagram RHR Heat Exchanger Bypass Valve E1150-F048B	T
6I721-2201-30	Schematic Diagram Service Water Bypass to RHR Valves E1150F073 & E1150F075	J
6I721-2201-33	Schematic Diagram RHR Service Water Pump A E1151C001A	V
6I721-2201-34	Schematic Diagram RHR Service Water Pump C E1151C001C	W
6I721-2201-35	Schematic Diagram RHR Service Water Pump B E1151C001B	P
6I721-2201-36	Schematic Diagram RHR Service Water Pump D E1151C001D	S
6I721-2201-37	Schematic Diagram Reactor Recirculation Extraction/Isolation to RHR Bypass Valve E11-F608	O
6I721-2201-38A	Schematic Diagram RHR Recirculation Outboard Bypass Valves E1150-F611A & E11-F611B	F
6I721-2201-74	Schematic Diagram RHR Heat Exchanger A Outlet Valve E1150F003A	B
6I721-2201-75	Schematic Diagram Suppression Pool to Pump C Valve E1150F004C	D
6I721-2201-76	Schematic Diagram Recirculation Line to Pump C Valve E1150-F006C	B
6I721-2201-77	Schematic Diagram Containment Spray Outboard Isolation Valve E1150F016A	G

Drawings

Number	Title	Rev or Date
6I721-2201-78	Schematic Diagram RHR Heat Exchanger A Inlet Valve E1150F047A	B
6I721-2201-79	Schematic Diagram Suppression Pool to Pump A Valve E1150F004A	F
6I721-2201-80	Schematic Diagram RHR Loop A Recirculation Inboard Isolation Valve E1150F015A	G
6I721-2201-81	Schematic Diagram RHR Loop A Recirculation Outboard Isolation Valve E1150F017A	F
6I721-2205-02	Schematic Diagram RHR Relay Logic A Circuit Part 1	X
6I721-2205-03	Schematic Diagram RHR Relay Logic A Circuit Part 2	Q
6I721-2205-04	Schematic Diagram RHR Relay Logic A Circuit Part 3	P
6I721-2205-05	Schematic Diagram RHR Relay Logic B Circuit Part 1	W
6I721-2205-06	Schematic Diagram RHR Relay Logic B Circuit Part 2	L
6I721-2205-07	Schematic Diagram RHR Relay Logic B Circuit Part 3	N
6I721-2205-09	Schematic Diagram RHR Instrumentation Circuit Part 2	P
6I721-2205-10	Schematic Diagram RHR Instrumentation Circuit Part 3	M
6I721-2205-11	Schematic Diagram RHR Power Distribution A & B Circuits	G
6I721-2205-12	Schematic Diagram RHR Annunciator Circuit Part 1	P
6I721-2205-13	Schematic Diagram RHR Annunciator Circuit Part 2	M
6I721-2205-14	Schematic Diagram RHR Annunciator Circuit Part 3	E
6I721-2205-15	Schematic Diagram RHR to Recirculation Loop Inboard Check & Locked Open Hand Valves E1100F050A & B, E1100F060A & B	N
6I721-2205-17	Schematic Diagram RHRSW Crosstie & Sample Line Valves. E11-F078, E11-F079A & B, E11-F080A & B	P

Drawings

Number	Title	Rev or Date
6I721-2205-21	Schematic Diagram For Bypass Valves E11-F610A & E11-F610B	E
6I721-2282-47	Internal-External Wiring Diagram Division II ECCS Trip Unit Cabinet H21-P083	J
6I721-2282-48	Internal-External Wiring Diagram Division II ECCS Trip Unit Cabinet H21-P083	E
6I721-2282-49	Internal-External Wiring Diagram Division II ECCS Trip Unit Cabinet H21-P083	C1
6I721-2282-50	Internal-External Wiring Diagram Division II ECCS Trip Unit Cabinet H21-P083	0
6I721N-2201-33	RHRSW Pump A E1151C001A	V
6I721N-2201-34	RHRSW Pump C E1151C001C	W
6I721N-2201-35	RHRSW Pump B E1151C001B	P
6I721N-2201-36	RHRSW Pump D E1151C001D	S
6M721-2006	Condensate Storage and Transfer System Process & Instrumentation Diagram	AY
6M721-2083	RHR Division II Process & Instrumentation Diagram	BD
6M721-2084	RHR Division I Process & Instrumentation Diagram	BA
6M721-2298-1	Piping Isometric Drywell RHR Return Line Division I	U
6M721-3035-1	Piping Isometric RHR Head Spray From Return Header to Drywell Penetration	W
6M721-3146-1	Piping Isometric RHR (North) Return from RHR Heat Exchanger to Drywell Penetration X13B	W
6M721-3151-1	Piping Isometric RHR (South) From RHR Heat Exchanger to Drywell Penetration X13A	AB
6M721-3152-1	Piping Isometric RHR Supply Header From Reactor Recirculation System	S
6M721-3154-1	Piping Isometric North RHR Pump Suction From Suppression Chamber	U
6M721-3157-1	Piping Isometric RHR Pump Discharge North	P
6M721-3158-1	Piping Isometric RHR Pumps Discharge to Heat Exchangers	R

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Number	Title	Rev or Date
6M721-3159-1	Piping Isometric RHR Containment Spray (North) From Return Header to Drywell Penetration	S
6M721-3160-1	Piping Isometric RHR Test Line & Suppression Chamber Spray Header	Y
6M721-3178-1	Piping Isometric RHR Minimum Flow Bypass (North) From Pump Discharge Header to Test Line	M
6M721-3184-1	Piping Isometric RHR Service Water Supply and Return To Heat Exchanger (South) Reactor Building Unit #2	W
6M721-3185-1	Piping Isometric Service Water Supply and Return From Row A to RHR Heat Exchanger (North)	T
6M721-3326-1	Piping Isometric Condensate from Condensate Storage Tank	H
6M721-3328-2	Piping Isometric Normal & Jockey Hotwell Supply Pump Discharge Piping	D
6M721-4217-1	RHRSW Supply Division I RHR Complex	J
6M721-4218-1	RHRSW Return Division I RHR Complex	L
6M721-4229-1	Upgrade Misc Piping Inside Torus From Quality Group D to D+	I
6M721-4598-1	Piping Isometric Demineralizer Water Supply Line to RHR (Division I) Return Header	B
6M721-5706-1	RHR Division II Functional Operating Sketch	X
6M721-5706-2	RHR Division I Functional Operating Sketch	V
6SD721-2500-01	One Line Diagram Plant 4160V & 480V System Service Unit-2	AA
6SD721-2500-02	One Line Diagram 13.8KV	G
6SD721-2500-03	One Line Diagram 4160V System Service Buses 64B, 64C	N
6SD721-2500-04	One Line Diagram 4160V System Service Buses #65E, 65F, 65G	O
6SD721-2500-05	One Line Diagram 4160V System Service Buses 64A, 65D, 65L Radwaste Building	R

Drawings

Number	Title	Rev or Date
6SD721-2500-06	One Line Diagram 4160V System Service Buses #66H, 69J, 68K, Pump Houses	P
6SD721-2500-08	One Line Diagram 4160V Diesel Gen. Buses #11EA, 12EB, & 14ED	L
6SD721-2500-09	Phasing Diagram Main Power System	K
6SD721-2500-10	Phasing Diagram 4160V & 480V System Service	G
6SD721-2500-11	Phasing Diagram 4160V & 480V System Service	A
6SD721-2530-01	One Line Diagram 120 VAC Instrument & Control Power Feeders Division 1 & 2 Reactor Building	R
6SD721-2530-11	260/130V 260V DC Battery Distribution 2PB Distribution - Division 2	AC
6SD721-2600-01	One Line Diagram 120 VAC Distribution Panels Division I & II Auxiliary Building - Relay Room Unit #2	P
6SD721-2600-02	One Line Diagram 120 VAC Distribution Panels BOP Auxiliary Building - Relay & Control Room	Y
6SD721-2600-03	One Line Diagram 120 VAC Distribution Panels Division I & II - Local Panels Unit #2	M
70756-A	Setting Plan of RHR Exchanger	4
70756-B	Shell Details & Assembly RHR Exchanger	2
70756-G	Supports and Lifting Lugs of RHR Exchanger	1
70756-H1	Cross-section (Assembly) of RHR Exchanger	1

Environmental Qualification Analyses

Number	Title	Revision or Date
EQ0-EF2-018	Summary of Environmental Parameters Used for Fermi 2 Environmental Qualification Program	H
EQ1-EF2-044	Valve Operator - Outside Containment with AC Motor	C
EQ1-EF2-052A	Solenoid Valve	D
EQ1-EF2-057	Valve Operator - Inside/Outside Containment	C
EQ1-EF2-058	RHR, & Core Spray Pump Motors	D
EQ1-EF2-063	Engineering Analysis for Rosemount Pressure Transmitters - Model 1153, Series B for Fermi 2	C

Environmental Qualification Analyses

Number	Title	Revision or Date
EQ1-EF2-237	Engineering Analysis for ITT Barton Differential Pressure Switch Models 580A & 581A	D

Generic Communications

Number	Title	Revision or Date
Bulletin 88-04	Potential Safety-Related Pump Loss	May 8, 1988
GL 97-04	Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps	October 7, 1997
GL 98-04	Potential for Degradation of the ECCS System and the Containment Spray System after a LOCA Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment	July 14, 1998
IN 87-10	Potential for Water Hammer During Restart of Residual Heat Removal Pumps	February 11, 1987
IN 87-10	Supplement 1	May 15, 1997
IN 96-55	Inadequate Net Positive Suction Head of Emergency Core Cooling and Containment Heat Removal Pumps under Design Basis Accident Conditions	October 22, 1996
IN 98-40	Design Deficiencies can lead to Reduced ECCS Pump Net Positive Suction Head During Design-Basis Accidents	October 26, 1998
IN 00-08	Inadequate Assessment of the Effect of Differential Temperature on Safety-related Pumps	May 14, 2000
NUREG CR-2772	ARL-398A, SAND82-7064, Hydraulic Performance of Pump Suction Inlets for Emergency Core Cooling Systems in Boiling Water Reactors	June 1982
NUREG 0661	Safety Evaluation Report Mark I Containment Long-Term Program and Supplement 1	August 1982
NUREG 0798	Safety Evaluation Report Related to the Operation of Enrico Fermi Atomic Power Plant, Unit No. 2, and Supplements 1-6	July 1981
NUREG 0897	Containment Emergency Sump Performance	April 1983

Generic Communications

Number	Title	Revision or Date
RG 1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps	November 2, 1970
RG 1.82	Water Sources for Long-Term Recirculation Cooling Following a LOCA	1

Lesson Plans

Number	Title	Revision or Date
LP-IC-331-1101	RHR Systems Instrument & Control Lesson Plan	4
ST-OP-315-0041-001	Fermi 2 Nuclear Training - Operations - RHR	January 2000

Modifications

Number	Title	Revision or Date
TCEDP 1404	Replacement of Transmitter E11N015A & B RHR System	B
TCEDP 7675	Revise the Opening Circuit of the RHR System Valves Such That When Valves Reach the Full Open Position the Valve Motor is Stopped by the Opening Limit Switch	B
TCTSR 23610	Addition of Motor Operated Valve for Containment Isolation of the Reactor Water Cleanup Return Line	0
TCTSR 26039	Modify RHR Pump Restart Logic Following Loss of Power Trip	0
TCTSR 27398	Evaluate/Modify Instrument Calibration Data Sheets for ERIS Points.	A
TCTSR 27727	Obsolete Fisher Controls Pressure Regulator (67FR-224) Replacement	0
TCTSR 27728	Obsolete Fisher Controls Pressure Regulator (67FR-221) Replacement	0
TCTSR 27759	GE Has Changed Relay Part Number from 145C3041P002 to DA148C6427P001. It is made from the same GE CR120K22041AA Relay with the Same Fit, Form and Function.	0

Modifications

Number	Title	Revision or Date
TCTSR 28108	Equivalent Part Identification - Agastat Control Relay Series EGP/ETR/EML Configuration Code Change from 003 to 004	C
TCTSR 28180	RHR Complex Pumps Freeze Protection	0
TCTSR 29384	RHRSW Minimum Flow Valve Controller Setpoint Discrepancies	0
TCTSR 29568	RHRSW Minimum Flow Valve - Design Specification and Design Basis Document Have Incorrect Setpoint	0
TCTSR 29634	As Built Notice: Rosemount Master Trip Unit (MTU) Model 510DU245049 Was Replaced by Model 710DUOTT27000	0
TCTSR 29975	Dampening Adjustment Specification Requirement for E11N021A	0
TCTSR 30397	Replacement of RHR Pump C Motor	0
TCTSR 30418	Replacement Gauge for U.S. Gauge Co. P602 Pressure Indicator Gauge	0
TCTSR 30811	Rosemount Has Changed the C25 Capacitors in the model 510DU and 710DU Master Trip Units	0
TDTSR 28998	RHR Minimum Flow Valve (E1150-F007B) Strokes Open Upon Initiation of Shutdown Cooling	0
TDTSR 30358	Impact of Reactor Building Temperature Increase on Instrument Loop Design Calculations	0

Operability Evaluations

Number	Title	Revision or Date
94-00328	Inadequate Pick-up Voltage Testing of RHR Time Delay Relays	September 12, 1994
97-13345	Appropriate RPV Level for Operation of RHR in Shutdown Cooling Mode Is Contradictory and Confusing	November 17, 1997
98-18661	Division 1 RHR Heat Exchanger Flow Rate Exceeded 10,700 gpm by Indication on Temporary Flow Meter	September 25, 1998
98-16038	Thermal Overloads on New Motor Control Center Positions May Not Have Been Tested Sufficiently to Satisfy Technical Specifications	July 16, 1998

Pre-Operational Tests (completed)

Number	Title	Revision or Date
PRET.E1100.001	RHR System, Sections 6.12 and 6.13	3
	Supplemental Test 19	September 24, 1984

Procedures

Number	Title	Revision or Date
20.205.01	Loss of Shutdown Cooling Abnormal Operating Procedure	17
20.300.SBO	Loss of Offsite and Onsite Power Abnormal Operating Procedure	0
23.205	RHR System Operating Procedure	72
23.208	RHR Complex Service Water Systems System Operating Procedure	65
24.321.07	480V Swing Bus 72 CF Automatic Throwover Scheme Operability	10
35.304.010	Refurbishing 5 HK Air Circuit Breakers	1
35.318.006	IAC53A,/B and IAC66B Overcurrent Relays	25
35.318.014	Medium Voltage Switchgear Breaker and Relay Control	28
35.318.015	PJC 11AV and PJC 12D Overcurrent Relay Testing	22
35.318.017	Inspection and Testing of Multi-Contact Auxiliary Relays	33
35.328.002	Electrical Scheme Checkout	28
35.329.001	Winding and Insulation Resistance Checks of 4160 AC Motors	22
42.000.02	Thermal Overload Relay Calibration	33
42.302.10	Calibration and Functional Test of Division II 4160 Volt Bus 65F Undervoltage Relays	28
42.309.03	Division I 18 Month 130/260 Voltage DC Battery Check	33
44.030.051	ECCS – RHR (LPCI Mode) Division I Logic Functional Test	35
44.030.052	ECCS – RHR (LPCI Mode) Division II Logic Functional Test	36

Procedures

Number	Title	Revision or Date
47.632.05	ECCS – RHR (LPCI Mode) Keep Fill, Division I and Division II, Channel Calibration	1
ARP 2D86	RHR Division I and II Fill Line Pressure Low Annunciator Response Procedure	7
IPM05	Post Maintenance Event Implementation and Feedback	2
MMA12	Instrument & Control Surveillance Test Setpoint Trending	1

Specifications

Number	Title	Revision or Date
22A1341AM	RHR Design Specification Data Sheet	7
3071-128	Electrical Design Instruction - Power Cable Selection	C
3071-128-EZ-01	Power and Control Power Fuse Sizing	B
3071-128-EZ-04	Electrical Design Instruction for Control and Power Cable Ampacities	B
C1-4180	Setpoint Validation Guidelines	B
T-3869	Pump Curve, RHR Pump B	February 1973
TC-3905	Pump Curve, RHR Pump D	February 1973

Surveillances

Number	Title	Revision or Date
24.204.01	Division I LPCI & Torus Cooling/ Spray Pump And Valve Operability Test	February 7, 2001
24.204.06	Division II LPCI & Torus Cooling/ Spray Pump And Valve Operability Test	March 21, 2001
44.020.304	Reactor Pressure - Shutdown Cooling Cut in Permissive Interlock, Division II Calibration/ Functional	June 7, 2000
44.020.302	Reactor Pressure - Shutdown Cooling Cut in Permissive Interlock, Division II Functional Test	December 4, 2000

Technical Requirements

Number	Title	Revision or Date
3.3.5.1	Emergency Core Cooling System Instrumentation	31
3.6.8	Drywell Spray	31

Technical Specifications

Number	Title	Revision or Date
3.3.5.1	ECCS Instrumentation	134
3.4.8	RHR Shutdown Cooling System - Hot Shutdown	134
3.4.9	RHR Shutdown Cooling System - Cold Shutdown	134
3.5	ECCS and Reactor Core Isolation Cooling System	134
3.5.1	ECCS – Operating	134
3.5.2	ECCS – Shutdown	134
B.3.5.1	ECCS – Operating Basis	0
B.3.5.2	ECCS – Shutdown Basis	0

Updated Final Safety Analysis Report Sections

Number	Title	Revision or Date
3.6.2	Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping Outside Containment	10
5.5.7	RHR System	10
6.2.2	Primary Containment Heat Removal System	10
6.3	Emergency Core Cooling Systems	10
7.1.2.1.3	ECCS Design Bases	10
7.1.2.1.27	Reactor Shutdown Cooling System Design Bases	10
7.3.1.2.4	Low Pressure Coolant Injection Instrumentation and Control	10
7.3.1.3	Analysis of the ECCS	10

Updated Final Safety Analysis Report Sections

Number	Title	Revision or Date
7.3.2.2.9	Containment and Reactor Vessel Isolation Control System Environmental Capabilities	10
7.4.1.3	Reactor Shutdown Cooling System Instrumentation and Control	10
7.4.2.4	Reactor Shutdown Cooling System Instrumentation and Control Analysis	10
7.5	Safety Related and Power Generation Display Instrumentation	10
15.6	Decreases in Reactor Coolant Inventory	10

Work Requests

Number	Title	Revision or Date
000Z002484	B31N6111B Analog Indicator Could Not Be Calibrated	August 1, 2000
000Z003750	Test Thermal Overloads	November 14, 2000
000Z934136	Modify Start Logic of RHR Pump B Per EDP-26039	May 19, 1994
000Z934137	Modify Start Logic of RHR Pump C Per EDP-26039	May 19, 1995
000Z934138	Modify Start Logic of RHR Pump D Per EDP-26039	May 19, 1994
0263000318	Perform 24.204.003 Division I & II LPCI Simulated Auto Actuation Test and Valve Operability Test	May 1, 2000
0777000502	Perform 44.030.295 ECCS Drywell Pressure - ADS Actuation, Division I, Channel A, Calibration/Functional	August 9, 2000
0790010426	Perform 47.632.05 ECCS RHR (LPCI Mode) Keep Fill Calibration, Division I & Division II	April 25, 2001
0790991028	Perform 47.632.05 ECCS RHR (LPCI Mode) Keep Fill Calibration Division I & Division II	October 29, 1999
0791000318	Perform 44.030.051 ECCS-RHR (LPCI Mode) Division I Logic Functional Test & Valve Actuation	May 2, 2000

Work Requests

Number	Title	Revision or Date
0792000318	Perform 44.030.052 ECCS-RHR (LPCI Mode) Division II, Logic Functional Test & Valve Actuation	April 19, 2000
2288000825	Perform 42.302.10 Division 2 Bus 65F 4160V Undervoltage Relay Technical Specification Calibration and Functional Test	November 09, 2000
E106000100	Megger Motor & Perform Polarization Index Test By NE-6.6-EQMS.036	May 08, 2000
E222940712	Perform Test on Medium Voltage Breaker and Relay Control, PI Motor	July 09, 1997
E230970804	Perform Test on Medium Voltage Breaker and Relay Control, PI Motor	May 08, 2000
E238950825	Perform Test on Medium Voltage Breaker and Relay Control, PI Motor	July 09, 1997
E246970804	Perform Test on Medium Voltage Breaker and Relay Control, PI Motor	May 08, 2000