January 13, 2006

Mr. D. Cobb,
Manager of Nuclear Generation
Detroit Edison Company
6400 North Dixie Highway
Newport, MI 48166

SUBJECT: FERMI POWER PLANT, UNIT 2
NRC BIENNIAL SAFETY SYSTEM DESIGN AND PERFORMANCE
CAPABILITY INSPECTION REPORT 05000341/2005016

Dear Mr. Cobb:

On December 2, 2005, the U.S. Nuclear Regulatory Commission (NRC) completed a baseline inspection at your Fermi Power Plant, Unit 2. The enclosed report documents the results of this inspection discussed on December 2, 2005, with Mr. William O'Connor, Jr. and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and to compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, performed walkdowns of equipment, observed activities, and interviewed personnel. This inspection specifically focused on the Reactor Core Isolation Cooling System and the Emergency Diesel Generators and associated support systems.

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

If you contest the subject or severity of a Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspectors' Office at the Fermi 2 facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Ann Marie Stone, Chief
Engineering Branch 2
Division of Reactor Safety

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

Enclosure: Inspection Report 05000341/2005016
w/Attachment: Supplemental Information

cc w/encl: N. Peterson, Manager, Nuclear Licensing
D. Pettinari, Legal Department
Compliance Supervisor
G. White, Michigan Public Service Commission
L. Brandon, Michigan Department of Environmental Quality -
Waste and Hazardous Materials Division
Monroe County, Emergency Management Division
Planning Manager, 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:	DPR-43
Report No:	05000341/2005016
Licensee:	Detroit Edison Company
Facility:	Fermi Power Plant, Unit 2
Location:	Newport, MI 48166
Dates:	November 14, 2005, through December 2, 2005
Inspectors:	A. Walker, Senior Engineering Inspector, Lead A. Klett, Engineering Inspector D. Schrum, Engineering Inspector R. Winter, Engineering Inspector M. Shlyamberg, Contractor G. Skinner, Contractor
Approved by:	Ann Marie Stone, Chief Engineering Branch 2 Division of Reactor Safety (DRS)

Enclosure

SUMMARY OF FINDINGS

IR 05000341/2005016(DRS); 11/14/2005 - 12/02/2005; Fermi Power Plant, Unit 2; Safety System Design and Performance Capability.

The inspection consisted of a review of the Safety System Design and Performance Capability (SSDPC) of the Reactor Core Isolation Cooling System and the Emergency Diesel Generators and associated support systems. The inspection was conducted by regional engineering inspectors. Three Green Non-Cited Violations, and three Unresolved Items (URIs) were identified. 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 may be Green, or assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to consider the effects of frequency variation on diesel generator loading. Specifically, the licensee's diesel generator loading calculations failed to account for increased loading that could result from allowable frequency variations above the nominal generator frequency of 60 Hz. The licensee's corrective action was to evaluate the need for revised margin in the calculation due to frequency variations.

  This issue was more than minor because it affected the Mitigating Systems Cornerstone objective of ensuring availability, reliability, and capability of systems needed to respond to a DB accident by failing to assure that the diesel generators would not inadvertently become overloaded. This finding was of very low safety significance because it screened out as Green using the SDP Phase 1 worksheet. (Section 1R21.2.b.1)

- Green. The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to translate the design basis requirements for each of the Emergency Diesel Generator starting air systems into specifications, procedures, and instructions. As a result of this failure, no objective evidence existed that the required emergency diesel generator starting air system capacity was being maintained. The licensee's corrective actions were to develop a formal calculation to document the acceptability of the Technical Specifications limit for the air capacity and to implement changes to the diesel starting air system and check valve testing, the process computer alarm setpoint, and the alarm response procedures.

  This issue was more than minor because it affected the Mitigating Systems Cornerstone objective of ensuring availability, reliability, and capability of systems needed to respond to a DB accident by failing to assure that the degradation of the capability of the diesel starting air system would be detected. This finding was of very low safety significance because it screened out as Green using the SDP Phase 1 worksheet. (Section 1R21.2.b.2)

- Green. The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the licensee’s failure to establish the correct condensate storage tank temperature limit for use in the plant accident analyses and net positive suction head calculations and for the failure to translate the condensate storage tank temperature limit into plant procedures to ensure that temperature limits are not exceeded. The licensee’s corrective action was the implementation of a tentative maximum condensate storage tank temperature limit and an analysis to demonstrate that there was adequate margin in the accident analysis.

  This issue was more than minor because it affected the Mitigating Systems Cornerstone objective of ensuring the reliability of Reactor Core Isolation Cooling, High Pressure Coolant Injection, and the Core Spray Systems because the failure to establish a temperature limit had the potential to reduce the margin of safety that the licensee believed to be available as a result of calculations. The finding was of very low safety significance because it screened out as Green using the SDP Phase 1 worksheet. (Section 1R21.2.b.3)

B. Licensee-Identified Violations

No findings of significance were identified.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstone: Mitigating Systems and Barrier Integrity

1R21 Safety System Design and Performance Capability (71111.21)

Inspection of safety system design and performance capability (SSDPC) verifies the initial design and subsequent modifications and provides monitoring of the capability of the selected systems to perform design basis functions. As plants age, the design basis may be lost and important design features may be altered or disabled. The plant risk assessment model is based on the capability of the as-built safety system to perform the intended safety functions successfully. This inspectable area verifies aspects of the Mitigating Systems and Barrier Integrity cornerstones for which there are no indicators to measure performance.

The objective of the SSDPC inspection is to assess the adequacy of calculations, analyses, other engineering documents, and operational and testing practices that were used to support the performance of the selected systems during normal, abnormal, and accident conditions. Specific documents reviewed during the inspection are listed in the attachment to the report.

The systems selected for inspection were the Reactor Core Isolation Cooling System (RCIC), the Emergency Diesel Generators (EDGs), and associated support systems. These systems were selected for review based upon the following criteria:

- having high probabilistic risk analysis rankings;
- having design basis overlap; and
- not having received recent NRC review.

The criteria used to determine the acceptability of the system's performance was found in the following documents:

- licensee Technical Specifications (TS);
- applicable Updated Final Safety Analysis Report (UFSAR) sections; and
- the systems' design documents.

.1 System Requirements

a. Inspection Scope

The inspectors reviewed the UFSAR, TS, system design basis documents, system descriptions, drawings, and other available design basis information, to determine the performance requirements of the RCIC system and the EDGs and associated support systems. The reviewed systems' 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 required review to ensure that the RCIC system was capable of providing cooling during Design basis events. The inspectors reviewed the net positive suction head calculations and verified alternate water source capacity. For the Emergency Diesel Generators and associated support systems, the inspectors verified the EDGs were capable of providing emergency power during Design basis events through review of design calculations.

Energy Sources: This attribute required review to ensure that the power source for major electrical equipment in the RCIC system was adequate for the proper functioning of the valves and other components. For the Emergency Diesel Generators and associated support systems, this attribute was reviewed by verifying power to required equipment. This review concentrated on EDG loading and reliability and battery sizing.

Controls: This attribute required review to ensure that required instrumentation calculations and surveillances for the RCIC and EDG systems were adequate. For both systems, the controls necessary for implementation of the necessary operating procedures and the swap over function of the condensate storage tank (CST) to the Torus upon low inventory of the CST were also reviewed.

Heat Removal: This attribute required a detailed review of the heat removal systems associated with the RCIC and EDG systems. The inspectors reviewed room cooling, lube oil cooling, and service water flow calculations.

b. Findings

No significant findings were identified. Two unresolved items (URIs) were identified.

b.1 Capability of RHR Complex Structure and Components to Withstand a Tornado

Introduction: The inspectors identified an Unresolved Item involving the lack of analyses to demonstrate the capability of the Residual Heat Removal (RHR) Complex structure and its enclosed components to withstand a tornado. This issue is unresolved, pending further review of Fermi's licensing basis by the NRC.

Description: The licensee was not able to provide an analysis or other documentation to demonstrate that the RHR Complex and its enclosed components were capable of withstanding the depressurization effects that could occur if a tornado passed directly over the building. The RHR Complex enclosed the EDGs, the EDG support systems, and ventilation systems. Each of the four EDGs was located in a separate room within the RHR Complex and was equipped with a ventilation system consisting of two supply fans, automatic modulating intake dampers, and a gravity exhaust damper.

The inspectors postulated that a tornado could pass directly over the RHR Complex. The intake dampers in each room would normally be closed when their associated EDG was not operating. If the EDG started (due to the tornado disabling offsite power), the intake dampers would remain closed, and the system would operate in a recirculation mode until the room temperature reached a predetermined level, at which time the intake dampers would fully or partially open. The gravity exhaust dampers appeared to provide an effective one way path for air to escape from the EDG rooms when the depressurization zone of a tornado would pass over the RHR Complex, thereby equalizing the air pressure inside the EDG rooms with the lower outside pressure. However, when the tornado depressurization zone would pass by the RHR Complex, the outside air pressure would be higher than the reduced pressure inside the diesel generator rooms, thereby closing the gravity exhaust dampers. The inspectors postulated that this phenomenon could result in a maximum pressure differential of 3 psid between the normal outside atmospheric pressure and the reduced inside atmospheric pressure. This differential pressure could develop across both the ventilation system intake and exhaust dampers as well as across the building structural components such as the roof.

The inspectors were concerned that the EDG support systems or the structure that enclosed the EDGs could be damaged as follows:

- First, the intake and exhaust dampers could be damaged and jammed so that they would not be able to perform their function when the EDGs started. The licensee indicated that the ventilation dampers were rated for approximately 6" of water plus seismic loads which were not specified. The approximately 3 psid postulated by the inspectors equated to 83" of water, which was considerably greater than the damper ratings.

- Second, the differential pressure across the roof would have added to the weight of the roof. The licensee was not able to demonstrate that this phenomenon had been analyzed.

- Finally, un-vented enclosures within the diesel rooms such as Motor Control Centers could be subjected to substantial forces due to the reduced ambient pressure within the rooms and the normal atmospheric pressure within the enclosures. The differential pressure could cause deformation of the enclosures, failed fasteners, and generation of missiles, which could degrade the enclosed equipment and other nearby equipment.

This issue was entered into the licensee's corrective action program as Condition Assessment Resolution Document (CARD) 05-26492 on November 17, 2005. The licensee's original corrective action was to perform a detailed analysis of the postulated tornado event to support the licensee's assertion that the RHR Complex was capable of withstanding the effects of a tornado. On December 15, 2005, the licensee updated this CARD to stop the development of this analysis. The licensee stated in the CARD that the specific issue of individual components located within the RHR Complex withstanding the effects of a tornado was not within Fermi 2's current licensing basis, and therefore, the calculation was not required.

The licensee stated in CARD 05-26492 that "Tornado design basis requirements were limited to the effect ... on safety related structures and the protection from tornado missile effects. Building internals are not subject to direct tornado differential pressure loading design and are protected by virtue of being housed within the Category I Structures." Section 2.3.1.3.2.2 of the Fermi 2 UFSAR stated, "Category I structures housing the systems required for a safe shutdown of the plant in the event of a tornado are designed to withstand the effects of a tornado by providing either sufficiently strong structures or appropriate venting. The design parameters of the Fermi 2 design basis tornado are: a) a rotational wind velocity of 300 mph, b) a translational wind velocity of 60 mph, c) an external pressure drop of 3 psi at the rate of 1 psi/sec."

The inspectors believed that an analyses to demonstrate that the RHR Complex structure, its ventilation systems and diesel generator support equipment would remain operable following the depressurization effects of a design basis tornado was necessary. The inspectors noted that concerns similar to those raised at the Fermi station had been recently raised at another nuclear station and were currently under review by the NRC for generic applicability. This issue is considered an unresolved item pending further review of Fermi 2's licensing basis by the NRC. (URI 05000341/2005016-01)

b.2 Potential Bypass of Secondary Containment via CST

Introduction: The inspectors identified an Unresolved Item concerning a potential radioactive release path via the CST following a loss of coolant accident (LOCA). This issue is unresolved pending clarification of the design basis with respect to the secondary containment bypass.

Description: The CST and the suppression pool serve as suction sources for the RCIC system. In standby condition, the system is normally aligned to the CST through a check valve E5150F011 (F011) and normally open motor operated valve (MOV) E5150F010 (F010). When the level in the CST decreases to a predetermined setpoint or when the level in the suppression pool increases to a predetermined setpoint, the suction path switches to the suppression pool as the normally closed MOVs E5150F029 (F029) and E5150F031 (F031) open and MOV F010 closes.

The HPCI system functions in a similar manner with CST suction check valve E4150F019 (F019) and normally open MOV E4150F004 (F004) and the suppression pool suction valves, normally closed MOVs E4150F041(F041) and E4150F042 (F042).

The inspectors were concerned that the licensee did not leak test the CST suction or the suppression pool suction valves. The inspectors postulated that following a design basis LOCA and a range of intermediate break LOCAs, the pressure differential between the suppression pool and CST could cause potentially contaminated, radioactive water to be transferred from the suppression pool to the CST through the MOVs and check valves.

The inspectors noted that the minimum (protected) CST level was approximately 592 feet, and the nominal suppression pool level was 557 feet. Under these circumstances, the CST could have a hydraulic overpressure of approximately 35 feet (592 feet minus 557 feet) or 15.14 psid over the nominal suppression pool level. The inspectors postulated that during an accident condition, the suppression pool level would rise due to a spill from the reactor coolant system break as the CST level decreases until the suction path is transfered to the suppression pool. This would result in a higher differential pressure (greater than 15.14 psid) between the suppression pool and CST. The inspectors noted that UFSAR Figure 6.2-11 illustrated a primary containment pressure response following a design basis LOCA. The graphical representation of data on this figure ended at approximately 6x104 seconds (16 hours and 40 minutes). The corresponding suppression pool (wetwell airspace) pressure at that time was approximately 18 psig. Because the licensee did not leak test the F011/MOV F010 and F019/MOV F004 pathways, the potential for leakage across these valves existed. Therefore, the increased pressure difference postulated during an accident scenario could result in the transfer of contaminated, radioactive water from the suppression pool to the CST. The radioactive iodine and other radioactive gases could come out of solution (Dalton's Law) once the solution reaches the CST, and escape into the environment. The amount of transfer and subsequent release would depend on an actual valve leakage and pressure differential. The release of radioactivity out of the CST could be outside of the current 10 CFR Part 100 and General Design Criteria 19 requirements. It should be noted that a failure of MOVs F010 and F004 to close could result in higher releases.

The licensee documented this issue as CARD 05-26699. The licensee believed that the secondary containment bypass leakage postulated in the above scenario was not part of plant design and licensing basis. The licensee based this position in part on the response to the Three Mile Island Question H.III.1.1.1, which stated that the CST was identified as isolated from highly contaminated systems. The licensee also stated that the plant design and licensing basis assumed ECCS liquid leakage occurred within the secondary containment boundary and was limited to a rate of 5 gpm. Furthermore, UFSAR Section 6.2.1.2.2.3 identified that the HPCI and RCIC CST suction lines were excluded as bypass leakage paths on the basis that they were sealed with water. The seal water is assumed to be the water in the HPCI and RCIC piping and in the CST. The inspectors concluded that the CST would be isolated from contaminated sources if the valves in question were shown to be leak-tight. Because this has not been demonstrated, the inspectors believed that licensee may not be meeting their licensing and design basis. Therefore, this issue is considered an unresolved item pending receipt of clarification of the design basis with respect to the secondary containment bypass. (URI 05000341/2005016-02)

.2 System Condition and Capability

a. Inspection Scope

The inspectors reviewed design basis documents, plant drawings, abnormal and emergency operating procedures, requirements, and commitments identified in the UFSAR and TS. The inspectors compared the information in these documents to applicable plant modifications and electrical, instrumentation and control, and mechanical calculations. The inspectors used applicable industry standards, such as the American Society of Mechanical Engineers (ASME) Code and the Institute of Electrical and Electronics Engineers, to evaluate acceptability of the systems' design. Select operating experience was reviewed to ensure issues were adequately evaluated and corrective actions were implemented, as necessary. The inspectors also reviewed operational procedures to verify that instructions to operators were consistent with design assumptions.

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

Installed Configuration: This attribute required detailed system walkdowns of the installed configuration of the RCIC System and the EDGs and associated support systems and their components necessary to perform the Normal Operating Procedures (NOPs) and Emergency Operating Procedures (EOPs). The walkdowns focused on the configuration of piping, components, and instruments as well as the environmental conditions in the areas and the potential vulnerabilities in regard to flooding and seismic events. The walkdowns also verified the installed configuration of components with design and licensing bases assumptions and design input values.

Operation: The inspectors verified that the RCIC system and the EDGs and associated support systems were consistent with design and licensing basis documents. Additionally, the inspectors verified that these design and license basis attributes were translated properly into the plant's operating procedures and EOPs.

Design: The inspectors reviewed the mechanical and electrical design of the RCIC System and the EDGs and associated support systems during design basis events to verify that the systems and subsystems would function as required. This included a review of the design basis, license basis, design assumptions, calculations, boundary conditions, and a review of selected modification packages.

Testing: The inspectors reviewed records of selected periodic testing and calibration procedures as well as surveillance procedures 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 that testing was consistent with design basis information.

b. Findings

Three findings of very low safety significance and one Unresolved Item were identified.

b.1 Non Conservative Calculation for Diesel Generator Loading

Introduction: The inspectors identified a finding involving a Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) for the licensee's failure to consider the effects of frequency variation on diesel generator loading. Specifically, the licensee's diesel generator loading calculations failed to account for increased loading that could result from allowable frequency variations above the nominal generator frequency of 60 Hz.

Description: The Surveillance test acceptance criteria given in Technical Specification 3.8.1 allowed diesel generator frequency to vary from its nominal value of 60 Hz by ±1.2 Hz (approximately 2 percent). This variance could cause the rotational speed of the connected induction motors to increase by approximately the same value (2 percent). Affinity laws for rotating equipment such as centrifugal pumps and fans showed that the equipment would do more work and consume more power as their rotational speeds increased. For instance, the hydraulic output power of a centrifugal pump would vary by the cube of its speed so that a 2 percent increase in speed would result in an approximately 8 percent increase in power consumed. The diesel generator loading during an emergency consisted predominantly of induction motors that could have increased speed as a result of frequency variations, and therefore could consume more power than the nominal values calculated at 60 Hz.

The Fermi Updated Final Safety Analysis Report (UFSAR) Section 8.3.1.1.8.3 stated, in part, "For all conditions calculated, the loads are within the short-time rating of the diesel generator in compliance with paragraph C.2 of Regulatory Guide 1.9, Revision 2." The short time (2 hour) rating of the diesel generators was given as 3135 kiloWatts (kW) in Fermi Design Calculation DC-5003, "Emergency Diesel Generator Loads Calculation," which calculated diesel generator loading. The inspectors noted that the automatically applied loads during the first ten minutes of an accident for Diesel Generator 14 were calculated to be 3133 kW versus the acceptance criteria of 3135 kW. The calculation had not considered additional loading that could result from frequency variations. Since there was only a 2 kW margin between the calculated and the permissible loading, the inspectors were concerned that allowable variations in frequency could cause inadvertent overloading of the diesels during an emergency. The inspectors noted, however, that recent surveillance test results showed diesel generator frequency was generally very close to the nominal value of 60 Hz, so that this did not present an immediate operability concern. The licensee entered this issue into their corrective action program as CARD 05-26681 to evaluate the need for revised margin in the calculation due to frequency variations.

Analysis: The inspectors concluded that the failure to perform adequate calculations to demonstrate that the diesel generators operated within their required ratings was a performance deficiency warranting a significance evaluation. This finding was considered more than minor in accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports" because it affected the Mitigating System Cornerstone objective of ensuring availability, reliability, and capability of systems needed to respond to a design basis accident. The licensee failed to assure that the diesel generators would not inadvertently become overloaded as a result of not considering the effects of frequency variation on diesel generator loading. In accordance with IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," Attachment 1, the inspectors performed an SDP Phase 1 screening by answering the questions in the Mitigating Systems Cornerstone Column. The inspectors determined that the finding was a design deficiency that was confirmed not to result in a loss of operability per "Part 9900, Technical Guidance, Operability Determination: Process for Operability and Functional Assessment." Therefore, the finding was determined to be of very low safety significance (Green).

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, Design Control requires that measures shall provide for verifying or checking the adequacy of design. Technical Specification 3.8.1 stated that the steady state frequency for the EDGs shall be between 58.89 Hz and 61.2 Hz. Contrary to this requirement, the licensee's calculations did not adequately verify or check the adequacy of diesel generator loading within the TS 3.8.1 limits. Specifically, calculation DC-5003 did not account for variations in diesel generator frequency that could cause loading in excess of the ratings of the diesel engines. Since this finding is of very low safety significance and was entered into the licensee's corrective action program as CARD 05-26681, it is considered a NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000341/2005016-03)

b.2 Adequate Leakage Criterion Not Established for the EDG Air Start System

Introduction: The inspectors identified a finding involving a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) for the licensee's failure to translate the design basis requirements for each of the Emergency Diesel Generator starting air systems into specifications, procedures, and instructions. Specifically, the licensee did not provide objective evidence that the required emergency diesel generator EDG starting air system capacity was being maintained without the aid of the air compressors.

Description: The inspectors reviewed the surveillance tests for the EDG air start system (Procedure 24.307.34/35/36/37; DGSW, DFOT and Starting Air Operability Test - EDG 11/12/13/14) and identified the following concerns:

1) The surveillance test measured the leakage of the check valve, which was the in-service test (IST) program requirement. The acceptance criterion required a final pressure in the accumulator to be equal to or greater than the TS surveillance 3.8.3.3 required value of 215 psig after about one minute following closure of the compressor control cutoff valves and compressor venting. The inspectors reviewed 19 completed surveillances and determined that in some cases, the accumulator pressure prior to isolation was as high as 255 psig. Therefore, a large leak rate of 40 psid per minute or over 15 percent per minute of pressure decay would result in a satisfactory test result of the IST check valve. The inspectors noted that TS Bases for the SR 3.8.3.3 stated, "this surveillance [verifying tank pressure is greater than 215 psig] ensures that, without the aid of the refill compressor, sufficient air start capacity for each EDG is available. The system design requirements provide for a minimum of five engine start cycles without recharging. The pressure specified in this SR is intended to reflect the lowest value at which the five starts can be accomplished." Although the surveillance test verifies pressure is greater than 215 psig, the inspectors determined that the test does not demonstrate that the air receivers are capable of holding that pressure without the aid of the compressors as stated in the TS bases.

2) In 2004, this surveillance procedure was modified to add a requirement to initiate a CARD if the pressure drop exceeded 10 psid. This pressure drop, however, was not an IST requirement nor a condition to fail the surveillance. Depending on the starting pressure of the tank, a pressure drop of 10 psid would result in EDG inoperability within a few seconds; however, this condition would result in a successful surveillance.

3) During the surveillance test, the normally open valve from each air receiver to a common shuttle valve on the compressor was isolated. The tubing downstream of the normally open valve and the remaining pressure retaining components

were non-safety related. By closing this normally open valve, air leakage from the non-safety related pressure retaining components was not monitored.

4) The accumulator pressure low-pressure alarm had a nominal set point of 220 psig. The alarm response procedure for this alarm (ARP 1A13-RHR, “Starting Air Pressure Low”) required the operators to start or verify that the air compressor was running. With a large leak, the pressure in the system would likely drop below the TS requirement prior to operator actions. In addition, depending on the time required to respond to this alarm and the actual leakage rate, the accumulator pressure could be lower than the minimum pressure required for a single start.

The licensee entered the above concerns into the licensee’s corrective action program as CARD 05-26451. The licensee’s corrective actions included changes to: (a) the EDG starting air system and check valve testing, (b) the process computer alarm set point, and (c) alarm response procedures.

Analysis: The inspectors concluded that the failure to establish adequate acceptance criterion for the system surveillance was a performance deficiency. This finding was considered more than minor in accordance with IMC 0612, “Power Reactor Inspection Reports,” because the finding could have affected the Mitigating System Cornerstone objective of ensuring availability, reliability, and capability of systems needed to respond to a design basis accident. Specifically, undetected degradation of the air receiver check could effected the starting capability of the EDGs.

In accordance with IMC 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations,” the inspectors performed an SDP Phase 1 screening by answering the questions in the Mitigating Systems Cornerstone Column. The inspectors determined that the EDGs remained operable because air receiver pressure was greater than the minimum TS requirement; therefore answered “no” to the five questions presented in the SDP worksheet. In addition, the inspectors could not postulate a credible scenario that could remove all 4 air starting compressors without a simultaneous start signal to the EDGs. Therefore, the finding was determined to be of very low safety significance (Green).

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, “Design Control” states, in part, that the design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Contrary to the above, the acceptance criteria for air receiver check valve leakage as stated in procedure 24.307.34/35/36/37; DGSW, DFOT and Starting Air Operability Test-EDG 11/12/13/14 was inadequate as it did not demonstrate that the required emergency diesel generator EDG starting air system capacity was being maintained without the aid of the air compressors. Because this finding is of very low safety significance and was entered into the licensee’s corrective action program as CARDs 05-26451 and 05-26642, it is considered an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000341/2005016-04) b.3 Failure to Translate CST Temperature Limit into Design Documents and Procedures

Introduction: The inspectors identified a finding of very low safety significance involving a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” having very low safety significance for the licensee’s failure to establish the correct CST temperature limit for use in the plant accident analyses and net positive suction head (NPSH) calculations and for the failure to translate this CST temperature into plant procedures to ensure that the temperature limit was not exceeded.

Description: During the inspection, the inspectors noted that although the CST temperature was monitored on operator logs, the licensee had not established a maximum temperature limit for the CST. The licensee assumed a CST temperature limit of 100 °F in their NPSH calculations. However, the inspectors noted that during the summer months, the CST temperature could be as high as 120 °F, due to recirculation of hotwell water.

Based on inspector’s concerns, the license issued CARD 05-26671 and also instituted a tentative maximum CST temperature limit of 120°F, which was the maximum design temperature for the CST structure. The licensee did an analysis to demonstrate that there was adequate margin in the accident analysis to keep the water above the top of the nuclear fuel at the higher CST temperature with a RCIC pump meeting its required water injection flow of 600 gallons per minute at 50 seconds.

The inspectors reviewed the input parameters to the NPSH calculation for the RCIC pump and determined that because of conservatism in other aspects of the calculation, the RCIC pumps would still have adequate NPSH to remain operable. The licensee had not completed an extent-of-condition review of all calculations, drawings, and inputs to the accident analyses assuming a new 120 °F temperature limit. On initial review, there appeared to be sufficient margins; however, these calculations would need to be revised for the new temperature limitation.

Analysis: The inspectors determined the failure to use the correct CST temperature in the plant accident analysis and NPSH calculations and the failure to translate the CST temperature limit into the station procedures was a performance deficiency warranting a significance evaluation. This issue was more than minor because it was associated with the Mitigating Systems Cornerstone attribute of Equipment Performance and affected the cornerstone objective of ensuring the reliability of the RCIC, HPCI, and Core Spray Systems. Specifically the inspectors determined that a higher initial CST temperature could lead to an increase in the suppression pool temperature, and a reduced NPSH for the RCIC, HPCI, and Core Spray Pumps. The failure to establish a temperature limit had the potential to reduce the margin of safety that the licensee believed to be available as a result of calculations.

In accordance with IMC 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations,” the inspectors performed an SDP Phase 1 screening by answering the questions in the Mitigating Systems Cornerstone Column. The inspectors determined that the finding was a design deficiency that was confirmed not to result in a loss of operability per “Part 9900, Technical Guidance, Operability

Determination: Process for Operability and Functional Assessment." There was significant margin in the accident analysis and NPSH to the pumps to account for the higher temperature in the CST. Therefore, the finding was determined to be of very low safety significance (Green).

Enforcement: 10 CFR Part 50 Appendix B, Criterion III, Design Control, requires, in part, that design control measures be established and implemented to assure that applicable regulatory requirements and the design basis for structures, systems, and components are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, the licensee had neither established the correct CST temperature limit for use in the plant accident analyses and NPSH calculations nor translated the CST temperature limit into plant procedures. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program, this finding is being treated as a non-cited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000341/2005016-05).

b.4 Inadequate Time Delay for Degraded Voltage Relays

Introduction: The inspectors identified an Unresolved Item regarding the adequacy of the time delay settings of the offsite power undervoltage relays. Specifically, the existing relay time delay settings were greater than the time allowed by the licensee's 10 CFR 50.46 LOCA sequential loading analysis.

Description: The inspectors reviewed the 4160V voltage system to assess vulnerabilities due to a potential loss of the preferred offsite source and the stand by onsite sources (emergency diesel generators). The inspectors evaluated the adequacy of the licensee's undervoltage protection scheme. During this review, the inspectors questioned the adequacy of the time delay settings of the offsite power undervoltage relays. Technical Specifications Table 3.3.8.1-1, "Loss of Power Instrumentation," listed the allowable values for the 4.16 kV emergency bus undervoltage (degraded voltage) relay time delays. These values were determined from Design Calculation DC-0919, "Undervoltage Relay Setpoints," and were listed in the TS as follows:

Bus Undervoltage Values:

Division I: $\geq$ 3873.0 V and $\leq$ 4031.0 V
Division II: $\geq$ 3873.0 V and $\leq$ 3776.0 V

Time Delay Values:

Division I: $\geq$ 41.8 seconds and $\leq$ 46.2 seconds
Division II: $\geq$ 20.33 seconds and $\leq$ 22.47 seconds

The duration of the undervoltage relay time delays would last from the detection of a sustained degraded voltage condition until the vital busses were transferred from offsite power to the EDGs. The Division I time delays of $\geq$41.8 seconds and $\leq$46.2 seconds were intended to allow sufficient time to start two RHR pumps and two Core Spray pumps and to allow the automatic load tap changer on transformer S.S. #64 to sufficiently improve voltage to prevent separation from the offsite source. The Division II time delays of $\geq$20.33 seconds and $\leq$22.47 seconds were intended to allow sufficient time to start two RHR pumps and two Core Spray pumps without causing separation from the offsite source.

The inspectors were concerned that the existing time delays were longer than the time allowed by the licensee’s 10 CFR 50.46 LOCA analysis sequential loading time of <13 seconds following receipt of a LOCA signal, as specified in Fermi 2 UFSAR Table 6.3-7.

Fermi 2 UFSAR Table 6.3-7, “Operation Sequence for Emergency Core Cooling Systems For Design Basis Accident,” listed the plant response (timing of the starting and loading of the EDGs and core cooling initiation) during the first ten minutes of a LOCA concurrent with a Loss of Offsite Power (LOOP). According to the Fermi 2 UFSAR, Section 8.3.1.1.8.1, “On occurrence of a LOCA and on receipt of an automatic signal from the power plant relays, each unit [EDG] automatically “fast-starts,” comes to rated voltage and synchronous speed, and is capable of operating as an isolated source to start the loads sequentially.... If a loss of system power has occurred, the EDG is automatically connected to the bus. If bus voltage is normal, the EDG stands by at synchronous speed and rated voltage.”

The inspectors postulated that a degraded voltage condition could exist concurrent with a LOCA Safety Injection Actuation Signal. The inspectors were concerned that if a degraded voltage condition existed concurrent with a LOCA, the voltage would be too low to power the ECCS equipment (motors) but high enough to prevent the EDGs from connecting to the safety-related buses. In addition, the undervoltage relay time delays would initiate which could delay the connection of the EDGs to the safety-related buses and thus delay low pressure core spray and coolant injection until the relay time delays timed-out (46.2 seconds for Division I and 22.47 seconds for Division II).

NRC Branch Technical Position PSB-1 Section B.1 stated that a second level of undervoltage protection should be provided with two separate time delays. Position B.1.b.1 required that the first time delay be of short duration (but longer than a motor starting transient) with a subsequent LOCA signal causing separation from the offsite source. The inspectors believed that in order to meet this requirement, the licensee’s degraded voltage scheme should have been capable of protecting safety-related equipment if a LOCA signal initiated at the same time that a degraded voltage condition existed. In addition, the inspectors reviewed an NRC letter dated June 2, 1977 (sent to all operating plants at that time) which stated that the allowable time delay for the degraded voltage protection scheme, including margin, “shall not exceed the maximum time delay that is assumed in the UFSAR accident analysis.”

The licensee was unable to demonstrate that during a LOCA concurrent with degraded voltage conditions, the 13-second time delay limit for the availability of power from the diesel generators could be met. During this delay, ECCS pumps may fail to start, and the MOVs may fail to move to their required positions, thus delaying water injection into the core. The licensee acknowledged the apparent discrepancy and initiated CARD 03-11847 to address the discrepancy. The resolution of CARD 03-11847 concluded that Fermi 2 was not committed to BTP PSB-1, Position B.1.b.1, citing various communications with the NRC. The inspectors did not note any exceptions to Position B.1.b.1 in these communications. NUREG-0798, issued on July, 1981, stated that the Fermi 2 undervoltage protection scheme "meets the Position 1 requirements and is acceptable." The inspectors believe BTP PSB-1 and the June 2, 1977 letter established the requirement for the degraded voltage scheme to respond to a degraded voltage condition concurrent with a LOCA.

The inspectors determined that applying a potentially non-conservative acceptance limit for the time delay relay did not assure the availability of the vital buses required to respond to an accident. The undervoltage relay time delay setpoint requirements needed appropriate evaluations and resolution of the design and licensing basis to ensure compliance with 10 CFR 50, General Design Criterion 17. This issue was initially raised during the Fermi 2003 SSDPC and documented as URI 05000341/2003007-02. The 2003 URI was closed in Inspection Report 05000341/2004004 based on the inspector's judgment that the licensee's planned corrective actions, as stated in CARD 03-11847 and which had not been implemented at the time of the inspector's follow-up activities, appeared to be adequate. After reviewing additional information during this inspection, the inspectors concluded that the resolution to CARD 03-11847 was inadequate and that the issue should be reopened as a separate URI in this inspection report. This issue is considered an unresolved item pending further NRC review to determine whether Fermi 2 is in compliance with their licensing basis for degraded voltage protection as provided in Branch Technical Position PSB-1. (URI 05000341/2005016-06)

.3 Components

a. Inspection Scope

The inspectors examined the RCIC System and the EDGs and associated support systems to ensure that component level attributes were satisfied. The inspectors specifically focused on EDG loading, the EDG service water pump and check valve, and the EDG fuel oil pump. The inspectors also reviewed the RCIC pump and the RCIC min flow, discharge and lube oil cooler MOVs. Specifically, the inspectors reviewed component degradation, component inputs/outputs, equipment and environmental qualification, equipment protection, and operating experience as described below.

Component Degradation: This attribute was reviewed to ensure that components were being maintained consistent with the design basis. The inspectors reviewed RCIC surveillance tests to ensure that equipment degradation, if present, was within allowable limits. Additionally, the inspectors performed a selective review to determine if the licensee was performing inservice testing in accordance with applicable requirements. Selected testing of the EDGs was also reviewed. Maintenance history was also reviewed for various components to ensure that there was not excessive degradation present.

Component Inputs/Outputs: The inspectors reviewed selected components in the RCIC and the EDG systems to ensure proper operation and input assumptions. Additionally, the inspectors verified selected component operation to ensure that the expected output/operation was consistent with desired outcomes.

Equipment/Environmental Qualification: This attribute was reviewed to ensure that the equipment was qualified to operate under the environment in which it is expected to be subjected to under normal and accident conditions.

Equipment Protection: The inspectors reviewed design information, specifications, and documentation to ensure that the RCIC and the EDG systems were adequately protected from those hazards identified in the UFSAR which could impact their ability to perform their safety function. Additionally, the inspectors verified adequate heating ventilation and air conditioning and freeze protection.

Operating Experience: This attribute ensures that applicable industry and site operating experience has been considered and applied to the components or systems. To verify this attribute, the inspectors reviewed licensee evaluations of operating experience including regulatory operability evaluations (OEs) and site OE (Corrective action documents and maintenance history) to ensure that the licensee had appropriately applied applicable insights to the systems and components reviewed.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA2 Problem Identification and Resolution

Review of Condition Reports

a. Inspection Scope

The inspectors reviewed a sample of RCIC System and the EDGs and associated support systems that were identified by the licensee and entered into the corrective action program. The inspectors reviewed these issues to verify an appropriate threshold for identifying issues and to evaluate the effectiveness of corrective actions related to design issues. In addition, issue reports written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problem into the corrective action program. The specific corrective action documents that were sampled and reviewed by the inspectors are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

Closed: (URI 05000341/2005014-02) Evaluation of EDG 12 High Bearing Temperature. An unresolved item was written on this item pending review of the RHR heating ventilation and air conditioning (HVAC) calculation and associated bearing temperature impacts. Specifically, the resident inspectors questioned the operability of EDG 12 for design conditions based on the outboard bearing temperature reaching 190° F during the 24-hour run. The issue was unresolved because the resident inspectors were concerned that the RHR HVAC may be ineffective at maintaining room temperatures. The resident inspectors were also concerned that at ambient temperatures above 80° F, the ability of RHR HVAC to keep system temperatures below the 195° F bearing shutdown limit and the required 202° F minimum viscosity limit was inconclusive.

During this inspection, the inspectors reviewed calculation DC-5489 and several CARDs written on bearing oil temperature concerns and concluded that room temperature had little effect on the bearing temperature, and that the generator load had the primary effect on the bearing temperature.

The inspectors did not identify any concerns with the licensee's calculations, or with the licensee's resolution of the resident inspectors' issue. Therefore, no performance deficiency or violation was identified, and this URI is closed.

4OA6 Meetings, Including Exits

.1 Exit Meeting

The inspectors presented the inspection results to Mr. W. O'Connor and other members of licensee management at the conclusion of the inspection on December 2, 2005. The licensee acknowledged the inspection results presented. The inspectors acknowledged that during the inspection some information provided by licensee personnel was identified as proprietary and will be treated appropriately. During the exit discussions no additional information was identified as proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

W. O'Conner, Vice President Nuclear Generation
D. Cobb, Director - Manager of Nuclear Generation
S. Berry, System Engineering Supervisor
E. Cavey, IST Engineer/Performance Engineer
R. Gaston, Licensing Manager
R. Johnson, Supervisor/Compliance
D. Kusumawati, Licensing Engineer
R. Libre, Director - Nuclear Engineering
A. Lim, Plant Support Engineering Supervisor
D. Noetzel, Manager - Plant Support Engineering
J. Pendergast, Principal Engineer/Licensing
P. Roelent, System Engineer, SDGs
S. Stasek, Director Nuclear Projects
L. Tremonti, System Engineering

Nuclear Regulatory Commission

C. Pederson, Director, Division of Reactor Safety
A. Stone, Chief, Engineering Branch 2, Division of Reactor Safety
M. Morris, Senior Resident Inspector
T. Steadham, Resident Inspector
L. Kozak, Senior Reactor Analyst

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000341/2005016-01	URI	Analyses to demonstrate the capability of RHR Complex Structure and Components to Withstand a Tornado not Available (Section 1R21.1.b.1)
05000341/2005016-02	URI	Potential Bypass of Secondary Containment via CST (Section 1R21.1.b.2)
05000341/2005016-06	URI	Inadequate Time Delay for Degraded Voltage Relays (Section 1R21.2.b.4)

Opened and Closed

05000341/2005016-03	NCV	Non Conservative Calculation for Diesel Generator Loading (Section 1R21.2.b.1)
05000341/2005016-04	NCV	Adequate Leakage Criterion Not Established for the EDG Air Start System (Section 1R21.2.b.2)
05000341/2005016-05	NCV	Failure to Translate CST Temperature Limit into Design Documents and Procedures (Section 1R21.2.b.3)

Closed

05000341/2005014-02	URI	Evaluation of EDG 12 High Bearing Temperature (Section 4OA5)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the Fermi 2 SSDPC inspection. This list includes documents prepared by others for the licensee. Inclusion on this list does not imply that NRC inspectors reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document in this list does not imply NRC acceptance of the document, unless specifically stated in the inspection report.

<u>1R21</u> <u>Safety System Design and Performance Capability</u>

<u>Engineering Changes</u>

EDP-8355; Modification of EDG Design Circuitry so that Neither ESA or ESB Relays will be defeated; Revision A

EDP 11501; Install Seismic Pins in RCIC Pump; dated December 20, 1990

EDP-13996; EDG Starting Air Receivers Low Pressure Alarm Pressure Switch Replacement; Revision 0

EDP 27431; Replace RCIC Turbine Sinspectors Inlet Isolation Valve E5150F045 with a Gate Valve; dated February 18, 1997

EDP-30815; EDG # 11,12,13,14 Starting Air System Piping and Valve Replacement; Revision 0

EDP-32366; Replace E5150F019, RCIC Pump Minimum Flow Motor Operated Valve with New Velan Motor Operated Gate Valve; Revision 0

EDP-33458; Revise DGSW Pump Min Flow Valve Setpoint; Revision A

EDP 3541; RCIC Governor Controls Jumper Installation and HPCI/RCIC Zenor Diode Replacement and Relocation; dated July 17, 1985

EDP 3641; Addition of Second Fuse for RCIC Barometric Condenser High Pressure Alarm Circuit; dated December 13, 1985

ERE-30993; Obsolete part replacement for EDG engine, air coolant control header relief valve; Revision A

TSR-27075; Colter has Supplied a Different Pump and Motor for the EDG Jacket Coolant Standesign basisy Pump

TSR-28543; Replacement of Valve E5150F045; Revision A

TSR-32614; Elimination of the Emergency Diesel Intercooler Keepwarm Function; Revision 0

Temp Mod 05-0022; Install a #12 AWG 1/C Ring Lugged Jumper Between Terminals 21 and 22 in MCC 2PA-1 Pos 6A; Revision 0

Temp Mod 05-0023; Install a #12 AWG 1/C Ring Lugged Jumper Between Terminals 18 and 20 in MCC 2PB-1 Pos 6C; Revision 0

10 CFR 50.59 Screenings and Evaluations

SE 90-0049; UFSAR Section 7.4.1.1.3.8 Revised to Clarify Description of Auto Transfer of RCIC/HPCI Suction From the CST to the Suppression Pool; Revision 0

SE 99-0007; TSR 30436 and LCR 99-016-UFS; Revision 0

50.59 Screen 05-0425; Installation of Jumper in MCC 2PB-1, pos 6C to Bypass Thermal Overload Trips in Control Circuit; dated September 23, 2005

50.59 Screen 05-0424; Installation of Jumper in MCC 2PA-1, pos 6C to Bypass Thermal Overload Trips in Control Circuit; dated September 23, 2005

SE 92-0026; ASME Section III Safety Relief Valves (SRVs) on the EDGs Air Receiver Tanks are being Replaced with ASME Section VIII Valves; dated August 25, 1992

Calculations

DC-4523; Reactor Wide Range Water Level Surveillance Procedure Validation; Revision H

DC-4531; CST Water Level; Revision B

DC-4561; RCIC Flow Measurement Calibration Surveillance Validation; Revision D

DC-5272; Sizing Criteria and Basis for Fuses Used In Power Distribution System and Control Circuits; Revision B

021-014-AW1; DC Motor Performance Calculation for Fermi DC Motor Operated Valves; Revision 1

DC-0213 Volume 1; Sizing of 130/260 V Batteries; Revision Q

DC-4943 Volume 1; DC Equipment Operability; Revision L

DC-5349 Volume 1; AC Control Cable Voltage Drop Calculation for QA1, Division I; Revision F

DC-5350 Volume 1; AC Control Cable Voltage Drop Calculation for QA1, Division II; Revision E

DC-5351 Volume 1; DC Control Cable Voltage Drop Calculation for QA-1, Division I; Revision E

DC-5719 Volume 1; Minimum Required Target Thrust (MRTT) for Generic Letter 89-10 Gate, Globe, and Quarter-turn Valves (torque); Revision K

DC-3225; Class 1E Equipment Qualification Review Reactor Core Isolation Cooling System; Revision D

DC-0502; RCIC Hydraulic Analysis; dated July 29, 1998

DC-4561; RCIC Flow Measurement Calibration Surveillance Validation; Revision D

DC-0885; ECCS Suction Line Air Ingestion; Revision C

DC-5426; High and Moderate Energy Line Break Evaluation; Revision C

DC-5779; Calculation of Mass and Energy Release Rates for High Energy Line Breaks Outside Containment; dated May 30, 2000

DC-4995; Assessment of I&C Equipment Operability in Dominant Areas of Concern During Station Blackout; Revision C

DC-4976; NUMARC Station Blackout Loss of Ventilation Effects on Temperature; Revision B

DC-4582; HPCI/RCIC Turbine Exhaust Pressure Monitoring; Revision B

DC-4591; RCIC/RWCU/HPCI Equipment Room Temperature Instrumentation Surveillance Procedure Validation; Revision F

DC-0446; Heat Loss Calculation for Condensate Storage Tank; Revision 0

DC-0106 Vol I; Cont Val Sizing & Selection RHR Sys; Revision D

DC-0548; Check Suction Head Reqm Diesel Day Tank RHR Complex; Revision A

DC-0502; RCIC Hydraulic Analysis; Revision D

DC-0551 Vol I; Sizing Emergency Relief Vent Diesel Fuel Oil Tanks Fuel Oil Storage Tank Emergency Relief Vents; Revision B,

DC-0566 Vol. I; Diesel Generator Service Water Pumps - RHR Complex; Revision C

DC-4531 Vol. I; CST Water Level; Revision B

DC-5474 VOL I; Water hammer Analysis for the RHR Service Water System; Revision A

DC-5804 Vol I and II; DGSW Design Basis Requirements; Revision C

DC-5950 Vol I; Design Basis System Parameters for AOVs R30F400, R30F401, R30F402 & R30F403; Revision A

DC-6191 Vol. I; Elimination Of Emergency Diesel Generator Intercooler Keepwarm Function; Revision 0

DC-5003, Emergency Diesel Generator Loads Calculation, Revision F

DC-0919, Undervoltage Relay Setpoints, Revision D

DC-5349, AC Control Cable Voltage Drop Calculation for QA1 Division I, Revision F

DC-5350, AC Control Cable Voltage Drop Calculation for QA1 Division II, Revision E

DC-6186, Electrical Loading, Short Circuit Current and Running Voltages for 4.16 kV 480 Volt and 120 Volt System Using ETAP PowerStation, Revision B

DC-4388, Protective Relay Settings for 13.2KV, 4.16KV, and 480 V Auxiliary Equipment, Revision E

Surveillances

23.702.01; Plant Systems Leak Check Procedure; Revision 11

24.206.01; RCIC System Pump and Valve Operability Test; Revision 63
Plant Technical Procedure 27.000.04; Freeze Protection Lineup Verification; Revision 31

24.206.04; RCIC System Automatic Actuation and Flow Test; Revision 44

42.309.01 DIV.1 & 2 Weekly 130/260 VDC Battery Check; dated November 10, 2005

42.309.02; DIV.1 Quarterly 130/260 VDC Battery Check; dated March 31, 2005

42.309.02; DIV.1 Quarterly 130/260 VDC Battery Check; dated June 30, 2005

42.309.02; DIV.1 Quarterly 130/260 VDC Battery Check; dated September 29, 2005

42.309.02; DIV.2 Quarterly 130/260 VDC Battery Check; dated March 17, 2005

42.309.02; DIV.2 Quarterly 130/260 VDC Battery Check; dated June 16, 2005

42.309.02; DIV.2 Quarterly 130/260 VDC Battery Check; dated September 15, 2005

42.309.03; DIV I 18 Month 130/260 VDC Battery Check (2A-1 Only); dated April 1, 2003

42.309.03; DIV I 18 Month 130/260 VDC Battery Check (2A-1 Only); dated November 13, 2004

42.309.03; DIV I 18 Month 130/260 VDC Battery Check (2A-2 Only); dated April 2, 2003

42.309.03; DIV I 18 Month 130/260 VDC Battery Check (2A-2 Only); dated November 16, 2004

42.309.05; DIV 1 (5 Year) 130/260 VDC Battery Check (2A-1 Only); dated November 7, 2001

42.309.05; DIV 1 (5 Year) 130/260 VDC Battery Check (2A-2 Only); dated November 9, 2001

44.030.258; ECCS - Reactor Water Level (Levels 1, 2 and 8) Division 2, Channel D Calibration Functional; Revision 48

44.030.258; ECCS - Reactor Water Level (Levels 1, 2 and 8) Division 2, Channel B Calibration Functional; Revision 43

44.030.401; ECCS - HPCI/RCIC Condensate Storage Tank Level Loop, E41-N061D Calibration Functional; Revision 23

44.110.004; Remote Shutdown RCIC Flow Indication Calibration; Revision 33

Plant Technical Procedure 27.000.07; Cold Weather Operations; Revision 0

Plant Technical Procedure 24.206.03; RCIC Discharge Piping Venting and Valve Verifications Test; Revision 38

Task ID; Perform 24.206.04 Sec-5.2 RCIC System Automatic Actuation Test; dated December 2, 2004

Task ID; Perform 24.206.01 RCIC System Pump Operability and Valve Test @ 1000 PSIG; dated October 30, 2005

Completed Surveillances

24.307.33, Perform 24.307.33 EDG No 14 24 Hour Run Followed by Hot Fast Start; dated September 16, 2005

24.307.33, Perform 24.307.33 EDG No 14 24 Hour Run Followed by Hot Fast Start; dated March 18, 2004

24.307.17, Perform 24.307.17 Sec-5.2 EDG Start and Load Test – Fast Start; dated April 15, 2005

24.206.01; RCIC System Pump and Valve Operability Test; Revision 63; dated December 03, 2004

24.206.01; RCIC System Pump and Valve Operability Test; Revision 63; dated March 04 2005

24.206.01; RCIC System Pump and Valve Operability Test; Revision 63; dated May 31 2005

24.206.01; RCIC System Pump and Valve Operability Test; Revision 63; dated August 30 2005

24.307.17; Emergency Diesel Generator 14 - Start and Load Test; Revision 46; dated January 28 2005

24.307.34; DGSW; DFOT and Starting Air Operability Test - EDG 11; Revision 44; dated October 27 2004

24.307.34; DGSW; DFOT and Starting Air Operability Test - EDG 11; Revision 45; dated

October 27, 2004

24.307.34; DGSW; DFOT and Starting Air Operability Test - EDG 11; Revision 45; dated January 26, 2005

24.307.34; DGSW; DFOT and Starting Air Operability Test - EDG 11; Revision 45; dated April 27, 2005

24.307.34; DGSW; DFOT and Starting Air Operability Test - EDG 11; Revision 45; dated July 27, 2005

24.307.34; DGSW; DFOT and Starting Air Operability Test - EDG 11; Revision 45; dated October 28, 2005

24.307.35; DGSW; DFOT and Starting Air Operability Test - EDG 12; Revision 47; dated February 25, 2005

24.307.35; DGSW; DFOT and Starting Air Operability Test - EDG 12; Revision 47; dated May 5, 2005

24.307.35; DGSW; DFOT and Starting Air Operability Test - EDG 12; Revision 47; dated August 4, 2005

24.307.35; DGSW; DFOT and Starting Air Operability Test - EDG 12; Revision 47; dated October 31, 2005

24.307.36; DGSW; DFOT and Starting Air Operability Test - EDG 13; Revision 44; dated January 29, 2005

24.307.36; DGSW; DFOT and Starting Air Operability Test - EDG 13; Revision 44; dated February 9, 2005

24.307.36; DGSW; DFOT and Starting Air Operability Test - EDG 13; Revision 44; dated May 11, 2005

24.307.36; DGSW; DFOT and Starting Air Operability Test - EDG 13; Revision 44; dated August 17, 2005

24.307.36; DGSW; DFOT and Starting Air Operability Test - EDG 13; Revision 44; dated November 9, 2005

24.307.37; DGSW; DFOT and Starting Air Operability Test - EDG 14; Revision 47; dated January 10, 2005

24.307.37; DGSW; DFOT and Starting Air Operability Test - EDG 14; Revision 47; dated February 16, 2005

24.307.37; DGSW; DFOT and Starting Air Operability Test - EDG 14; Revision 47; dated May 18, 2005

24.307.37; DGSW; DFOT and Starting Air Operability Test - EDG 14; Revision 47; dated August 12, 2005

24.307.49; EDG 11 DGSW Pump Discharge Check Valve Operability Test; Revision 4; dated May 1, 2002

24.307.49; EDG 14 DGSW Pump Discharge Check Valve Operability Test; Revision 4; dated June17, 2003

24.307.49; EDG 12 DGSW Pump Discharge Check Valve Operability Test; Revision 4; dated August 4, 2004

24.307.49; EDG 13 DGSW Pump Discharge Check Valve Operability Test; Revision 5; dated August 17, 2005

43.401.703; EDG-11, Service Water Min Flow Valve Stroke Time Test, Revision 4; dated April 6, 2003

43.401.703; EDG-11, Service Water Min Flow Valve Stroke Time Test, Revision 4; dated November 11, 2004

44.060.001; Offline RCIC System Logic Functional Test, dated June 6, 2003

Procedures

20.307.01; Emergency Diesel Generator Failure, Revision 21

23.307; Emergency Diesel Generator System, Revision 86

24.307.13; Emergency Diesel Generator 14 – ECCS Start and Load Rejection Test and Logic Functional Tests of Bus 65F Breakers, Revision 39

35.306.008; ITE Gould Motor Control Center Load Compartment; Revision 46

35.306.018; Spectrum Motor Control Center Load Compartment; Revision 4

35.306.005; MCC Bus and Compartment General Inspection and Maintenance; Revision 27

35.318.014; Medium Voltage Switchgear Breaker and Relay Control; Revision 32

35.304.010; Refurbishing 5 HK Air Circuit Breakers; Revision 7

35.306.001; 480 Volt Switchgear Breaker and Relay Control Testing; Revision 33

35.304.009; Refurbishing 600 Volt K-line Air Circuit Breakers; Revision 27

3071-128-EZ-03; Electrical Design Instructions Thermal Overload Sizing; Revision C

3071-128-EJ; Electrical Engineering Standard Class 1E, 1M & BOP Equipment Fusing Requirements; Revision BJ

Calibration Specification Sheet E5IN006-S; Pump Suction Low Pressure Trip; dated July 28, 2000

Setting Sheets

R30R560A-S; Calibration Specification Sheet Emergency Diesel Generator EDG 11 West Air Receiver Pressure Indication, Revision A

R30R560A-S; Calibration Specification Sheet Emergency Diesel Generator EDG #11 Division 1 Compressor R3000D001 Starting Air System Pressure Switch, Revision A

R30N560A-S; Calibration Specification Sheet Air Receiver R3000A011 Low Pressure Switch, Revision 0

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R30N555A-S; Diesel Fuel Oil Storage Tank R3000A001 High Level, Revision A

R30N556A-S; Diesel Fuel Oil Storage Tank R3000A001 Low Level Switch, Revision A

R30R570A-S; Emergency Diesel Generator EDG No. 11 Fuel Oil Day Tank Level, Revision 0

R30R563A-S; Diesel Fuel Oil Day R3000A017 Low Level Switch, Revision E

Condition Assessment Resolution Documents (CARDs) Reviewed During the Inspection

04-20679; RHR switchgear outside air damper or control logic not functioning

04-20879; X4103 RHR Complex Ventilation System Maintenance Rule (a)(1) status

05-22196; Broken C-clip on one of the damper blades where it attaches to the crank arm

05-22325; EDG 14 Switchgear Room Fans Run and Auto Lights Pulsating Bright to Dim

04-23427; Damper Linkage Disconnected Due to Loose Bolt

05-23568; Damper Linkage Found Incorrectly Installed on X4103F139 - EDG 14 Switchgear Outdoor Air Damper

05-25252; Blown Control Power Fuse

05-25355; EDG 12 engine room cooler than normal

05-25992; Develop a Design Calculation Reference and Subject Matter Experts (SMEs)

05-25993; Develop a Design Calculation Reference and Subject Matter Experts (SMEs)

05-25994; Develop a Design Calculation Reference and Subject Matter Experts (SMEs)

05-25995; Develop a Design Calculation Reference and Subject Matter Experts (SMEs)

05-25393; Various DC MOVs are possibly Inoperable Due to TOL Sizing Not Meeting the Requirements of 3071-128-EZ-03; dated September 22, 2005

05-23959; Potential Inadequate Coordination in Equipment used in the Dedicated Shutdown Scenario Investigation; dated June 30, 2005

05-25118; 62 Device (Time Delay Relay) Failed as Found; dated September, 6 2005

05-21463; High SWI Results on RCIC Pump Outboard Bearing; dated March 4, 2005

05-23416; RCIC Isolation Due to Failed Instrument; dated June 6, 2005

98 12269; Non-conservative Assessment of Worst Case DC MOV Stroke Times; dated April 1, 1998

97-11486; NRC Notice of Violation on Condensate Storage Tank Volume; dated October 31, 1997

01-14733; Potential RATS Commitment Non-compliance; dated May 5, 2001

03 18351; Engineering Functional Analysis to Support Removal of Floor Plugs During RCIC Outage; dated June 18, 2003

99-02412; RCIC Control Valve Not Going Full Open; dated November 7, 1999

CARDs Generated Due to the Inspection

05-26435; NRC Concern Associated with Barrier Analysis; dated November 16; 2005

05-26442; NRC Identified Issue SSDP-010 - Lack of Standard for Cotter Pin Installations; dated November 16; 2005

05-26451; NRC Concern - Procedure 24.307.34 Starting Air Operability Test - EDG 11 and Associated EDG Tests; dated November 16; 2005

05-26492; NRC SSDPC Issue - Design Calculation for RHR Complex Depressurization is not available; dated November 17; 2005

05-26496; Design Basis Calculations Could not be Found to Support the Basis for Level Switch Setpoint; dated November 18; 2005

05-26509; RCIC EGM Tuning Required; dated November 21; 2005

05-26522; DC-5349 Vol.1, DC-5350 Vol.1, DC-5351 Vol.1 and DC-5352 Vol.1 Methodology Concerns; dated November 21; 2005

05-26529; NRC Concern - Potential Inadequate Design Verification of EDG Auxiliary Process Cooling; dated November 21; 2005

05-26544; SSPD-047 - NRC Inspector asks if the EDG Muffler Exhaust can be Crimped During a Seismic Event; dated November 22; 2005

05-26554; SSPD-033 - At the NRC Inspector's Request Evaluate the Engagement for the EDG Governor Linkage Connector; dated November 23; 2005

05-26608; SSPD-033 - Missing Seismic Calculations; dated November 28; 2005

05-26618; Enhancement to DC-5474; dated November 28; 2005

05-26640; Incorrect Nominal Motor Voltage Shown in Calculation; dated November 29; 2005

05-26642; NRC identified issue: SSDP-068 - Concerns on pre-operational testing adequacy for the EDG Starting Air system; dated November 29; 2005

05-26666; Revise 35.307.004 to add governor shaft engagement criteria; dated November 30; 2005

05-26671; 2005 SSDI Item 108: Design Limits on CST Maximum Temperature Not Translated into Plant Procedures; dated December 1, 2005

05-26676; NRC Identified Issue - SSDP-128: Lack of programmatic testing on the RCIC CST Suction check valve; dated December 1, 2005

05-26677; NRC Identified Issue, SSPDI RFI #126, Concern with Reg Guide 1.9 Rev 2 compliance, in conjunction with Underfrequency Relay (81 device) and Bus Differential Relay (Device 87B) testing in accordance with SR 3.8.1.12.; dated December 1, 2005

05-26681; NRC identified concern during SSPDI inspection on RFI-123; dated December 1, 2005

05-26685; NRC Inspection Item SSDP-064; dated December 1, 2005

05-26690; Effect of CST water temperature on HPCI/RCIC Pumps Suction Swap; dated December 1, 2005

05-26699; NRC Identified Issue- Question Regarding Potential ECCS Secondary Containment Bypass Leakage Via CST; dated December 1, 2005

Drawings

5SD721-2530-13; Frontal Elevation 260V DC MCC 2PA-1 DIV 1 Auxiliary BLDG 3rd Floor; Revision AH

5SD721-2531-08; DC MCC Internal Diagram Reversing Full Voltage Starter With Remote Control Transfer; Revision J

6I721-2030-01; Logic Diagram Reactor Core Isolation Cooling System; Revision D

6I721-2030-06; Logic Diagram Reactor Core Isolation Cooling System; Revision D

6I721-2030-07; Logic Diagram Group 8 - RCIC Isolation Valve Mimic Display; Revision 0

6I721-2231-11; Schematic Diagram RCIC Sinspectors Inlet Bypass Valve E5150F095; Revision C
6I721-2235-01; Schematic Diagram RCIC System Notes, Relay Tabulation and Power Distribution; Revision V
6I721-2045-56; Internal & External Diagram RCIC Cabinet H11P621; Revision U

6I721-2231-03; Schematic Diagram RCIC STM Line Inboard Isol Vlv and Trip & Throttle Vlv. E5150F007 & E5150F059; Revision W

6I721-2231-05; Schematic Diagram RCIC Pump Disch Valves To FW Header E5150F012 & E5150F013; Revision V

6I721-2231-07; Schematic Diagram RCIC Turb. Sinspectors Inlet, Turb. Cooling Water Supply and Sinspectors Inlet Bypass Valves E5150F045 & F046; Revision X

6I721-2231-08; Schematic Diagram Min-Flow Bypass & Test Valves E5150F019 & E5150F022; Revision Y

6SD721-2530-10; One Line Diagram 260/130V Ess. Dual Battery 2PA Distribution – Division I; Revision AF

6SD721-2530-11; One Line Diagram 260/130V Ess. Dual Battery 2PB Distribution – Division II; Revision AE

6M721-2044; Diagram Reactor Core Isolation Cooling System (RCIC); Revision AY

6M721-2225; Floor & Equipment Drain Risers All Floors in Reactor Building; Revision W

6M721-2045; Reactor Core Isolation Cooling System Barometric Condenser; Revision AO

6I721-2611-41; Reactor Building Core Spray Emergency Equipment Cooler #1 & RCIC Cooler #2 Fans T4100B020 & T4100B021; Revision 0

6M721N-2050; P & ID Equipment Drains Division I and II - RHR Complex; Revision M

6M721-2006; Condensate Storage and Transfer System Diagram; Revision BB

6M721-2035; Diagram High Pressure Coolant Injection System (HPCI) Reactor Bldg; Revision BF

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6M721-2045; Diagram Reactor Core Isolation Cooling System (RCIC) Barometric Condensor; Revision AO

6M721N-2046; P & ID Diesel Generator System Div I RHR Complex; Revision AB

6M721N-2047; P & ID Diesel Generator Sys Div II RHR Complex; Revision AD

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6M721N-2049; P&ID Diesel Fuel Oil System & Lube Oil System Division II RHR Complex; Revision AK

6M721N-2052; P & ID RHR Service Water System Division 1 RHR Complex; Revision AC

6M721N-2053; P & ID RHR Service Water System Division 2 RHR Complex; Revision AE

6M721N-2056; P & ID Diesel Generator Ventilation RHR Complex Division I; Revision Q

6M721N-2057; P & ID Diesel Generator Ventilation RHR Complex Division I; Revision R

6M721N-2058; P & ID Diesel Generator Ventilation RHR Complex Division 2; Revision R

6M721N-2059; P & ID Diesel Generator Ventilation RHR Complex Division 2; Revision Q

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61721N-2572-20, Schematic Diagram 4160V ESS Diesel Bus 14ED Load Shedding Strings, Revision W

6I721-2215-02, Elementary Diagram Core Spray System Relay Logic Systems 1 & 2 and Relay Schedules, Revision AB

6I721-2230-01, Logic Diagram Reactor Core Isolation Cooling System, Revision D

6I721-2230-02, Logic Diagram Reactor Core Isolation Cooling System, Revision C

6I721-2230-04; Logic Diagram Reactor Core Isolation Cooling System, Revision C

6I721-2231-03; Schematic Diagram RCIC Stm Line Inboard Isol. VLV & Trip Throttle Vlv E5150F007 & E5150F059, Revision W

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6I721-2231-05; Schematic Diagram RCIC Pump Disch Valves to FW Header E5150F012 & E5150F013

6I721-2231-06; Schematic Diagram RCIC System Suppression Valves E5150F007 & E5150F059, Revision Q

6I721-2231-07; Schematic Diagram RCIC Turb. Sinspectors Inlet , Turb. Cooling Water Supply and Sinspectors Inlet Bypass Valves E5150F045 & F046, Revision X

6I721-2231-08; Schematic Diagram Min-Flow Bypass & Teas Valves E5150F019 & E5150F022, Revision Y

6I721-2235-01; Schematic Diagram RCIC System Notes, Relay Tabulation, and Power Distributions, Revision V

6I721-2235-02; Schematic Diagram RCIC System Logic Circuit Part 1, Revision S

6I721-2235-03; Schematic Diagram RCIC System Logic Circuit Part 2, Revision Z

6I721-2235-05; Schematic Diagram RCIC Turbine Governor and Trip and Throttle Valve and Remote Trip Circuit, Revision O

6I721-2572-11; Schematic Diagram 4160V ESS Diesel Bus 12EB Pos EB3, Revision W

6I721-2572-12; Schematic Diagram 4160V ESS Diesel Bus 12EB Pos EB5, Revision O

6I721N-2578-12; Relaying & Metering Diagram Diesel Generator #14, Revision Z

6I721N-2711-36; Schematic Diagram Diesel Generator #14 Control Part 1, Revision AA

6I721N-2711-37; Schematic Diagram Diesel Generator #14 Control Part 2, Revision AB

6I721N-2711-38; Schematic Diagram Diesel Generator #14 Annunciator System, Revision R

6I721N-2711-40; Schematic Diagram Exciter & Voltage Regulator and Governor Control Diesel Generator #14, Revision 14

6M721-5709-1; Reactor Core Isolation Cooling System Sketch - Functional Operating Sketch; Revision AF

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VMR1-50; General Electric Analog Trip Unit 184C5988G100-G799; Revision C

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VD-001; Ventilation Air Quantity for Diesel Generator Room #11; Revision 2

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List of CARDS for RCIC Valves (5 years); dated November 29, 2005

RCIC Turbine Maintenance History; dated November 30, 2005

RCIC Low Pressure Time Response Test; dated December 2, 2004

Plant Technical Procedure 23.206; Reactor Core Isolation Cooling System; Revision 82
Design Basis Document No. E51-00; Reactor Core Isolation Cooling; January 19, 2000
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IEEE Std 387-1977; IEEE Standard Criteria for Diesel Generator Units Applied as Standesign basisy Power Supplies for Nuclear Power Generating Stations

IEEE Std 450-1972; IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Stationary Type Power Plant and Substation Lead Storage Batteries; dated 1972

IEEE Std 484-2002; IEEE Recommended Practice for Installation Design and Installation of Vented Lead-Acid Batteries for Stationary Applications; dated 2003

IEEE Std 485-1997; IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications; dated 1997

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NRC Integrated Inspection Report 05000341/2004004, dated July 27, 2004

NRC Safety System Design and Performance Capability Inspection Report 50-341/03-07(DRS), dated July 30, 2003

NUREG-0798; Safety Evaluation Report related to the operation of Enrico Fermi Atomic Power Plant Unit No. 2, dated July 1981

Regulatory Guide 1.9; Selection, Design, and Qualification of Diesel-Generator Units Used as Standesign basisy (Onsite) Electric Power Systems at Nuclear Power Plants, Revision 2

Regulatory Guide 1.76; Design Basis Tornado for Nuclear Power Plants, dated April 1974

ISI-IST Program; Revision 5, Change 12

IST-STB; IST Program Summary Valve Stroke Time Basis; Revision 38

LIST OF ACRONYMS USED

AC	Alternating Current
ADAMS	Agencywide Documents Access and Management System
ASME	American Society of Mechanical Engineers
CARD	Condition Assessment Resolution Document
CFR	Code of Federal Regulations
CST	Condensate Storage Tank
DB	Design Basis
DER	Deviation Event Report
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedures
GDC	General Design Criteria
HVAC	Heating, Ventilation and Air Conditioning
IMC	Inspection Manual Chapter
ISI	In-Service Inspection
IST	In-Service Testing
KW	Kilo Watt
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
MOV	Motor Operated Valve
NCV	Non-Cited Violation
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
SDP	Significance Determination Process
SSDPC	Safety System Design and Performance Capability
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item