August 29, 2005

Mr. Christopher M. Crane
President and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2
COMBINED NRC BIENNIAL MODIFICATION, CHANGES, TESTS, AND EXPERIMENTS, AND SAFETY SYSTEM DESIGN AND PERFORMANCE CAPABILITY INSPECTION REPORT 05000373/2005008(DRS); 05000374/2005008(DRS)

Dear Mr. Crane:

On July 26, 2005, the U.S. Nuclear Regulatory Commission (NRC) completed a combined baseline inspection at your LaSalle County Station, Units 1 and 2. The enclosed report documents the results of this inspection discussed on July 26, 2005, with the Site Vice President, Ms. Susan Landahl, 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. The safety system design and performance capability portion of this inspection specifically focused on the 125/250 Volt Direct Current Distribution and Reactor Core Isolation Cooling systems.

Based on the results of this inspection, six 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 LaSalle County Station.

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 Nos. 50-373; 50-374
License Nos. NPF-11; NPF-18

Enclosure: Inspection Report 05000373/2005008(DRS); 05000374/2005008(DRS)
w/Attachment: Supplemental Information

cc w/encl: Site Vice President - LaSalle County Station
LaSalle County Station Plant Manager
Regulatory Assurance Manager - LaSalle County Station
Chief Operating Officer
Senior Vice President - Nuclear Services
Senior Vice President - Mid-West Regional Operating Group
Vice President - Mid-West Operations Support
Vice President - Licensing and Regulatory Affairs
Director Licensing - Mid-West Regional Operating Group
Manager Licensing - Clinton and LaSalle
Senior Counsel, Nuclear, Mid-West Regional Operating Group
Document Control Desk - Licensing
Assistant Attorney General
Illinois Emergency Management Agency
State Liaison Officer
Chairman, Illinois Commerce Commission

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Manager Licensing - Clinton and LaSalle
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Operating Group
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Assistant Attorney General
Illinois Emergency Management Agency
State Liaison Officer
Chairman, Illinois Commerce Commission

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-373; 50-374
License Nos:	NPF-11; NPF-18
Report No:	05000373/2005008(DRS); 05000374/2005008(DRS)
Licensee:	Exelon Generation Company, LLC
Facility:	LaSalle County Station, Units 1 and 2
Location:	2601 N. 21st Road Marseilles, IL 61341
Dates:	June 27, 2005 through July 26, 2005
Inspectors:	P. Lougheed, Senior Engineering Inspector, Lead J. Jacobson, Senior Engineering Inspector B. Daley, Senior Engineering Inspector D. Schrum, Engineering Inspector C. Acosta, Engineering Inspector M. Munir, Engineering Inspector
Approved by:	A. M. Stone, Chief Engineering Branch 2 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000373/2005008(DRS); 05000374/2005008(DRS); 06/27/2005 - 07/26/2005; LaSalle County Station; Units 1 and 2; Changes, Tests, and Experiments; and Safety System Design and Performance Capability.

The inspection was a combined baseline inspection of changes, tests, and experiments; permanent plant modifications; and the safety system design and performance capability of the reactor core isolation cooling and the 125/250 volt direct current distribution systems. The inspection was conducted by regional engineering inspectors. Six Green Non-Cited Violations 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 be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance was identified by the inspectors associated with a Non-Cited Violation of 10 CFR 50.59, "Changes, Tests, and Experiments," where the licensee failed to complete a full evaluation in accordance with 10 CFR 50.59 for an adverse change to the nitrogen supply header description in the updated final safety analysis report. This issue was entered into the licensee's corrective action system.

  This finding was more than minor because the screening was adverse and there was insufficient information to reasonably conclude that prior NRC approval was not necessary. This finding was categorized as Severity Level IV because the underlying technical issue for the finding was determined to be of very low safety significance using the Phase 1 worksheet. (Section 1R02.b)

- Green. A finding of very low safety significance was identified by the inspectors associated with a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," where the licensee failed to maintain an accurate design basis supporting the addition of loads on the safety-related buses, due to the simultaneous energization of both battery chargers. This issue was entered into the licensee's corrective action system and the licensee performed a preliminary analysis which showed that the safety-related buses would not be overloaded with both chargers energized simultaneously.

  This finding was more than minor because it affected an attribute of the mitigating systems cornerstone. Specifically, the licensee could not initially demonstrate that the design basis of the plant was not affected by adding the additional battery charger load. This finding was of very low safety significance because it screened out using the Phase 1 worksheet. (Section 1R21.1.b1)

- Green. A finding of very low safety significance was identified by the inspectors associated with a Non-Cited Violation of 10 CFR 50.63, “Loss of All Alternating Current Power.” Specifically, LaSalle County Station failed to maintain an accurate design basis heat-up calculation that supported the heat loads that would be present during a station blackout event for the water leg pump room. This issue was entered into the licensee’s corrective action system and the licensee performed a preliminary analysis which showed that the temperatures in the water leg pump room were within previously analyzed limits.

  This finding was more than minor because it affected an attribute of the mitigating systems cornerstone. Specifically, the licensee had not maintained design control over the maximum heatup temperature in the water leg pump room which are necessary for coping with a station blackout. This finding was of very low safety significance because it screened out using the Phase 1 worksheet. (Section 1R21.1.b2)

- Green. A finding of very low safety significance was identified by the inspectors associated with a Non-Cited Violation of 10 CFR 50.63, “Loss of All Alternating Current Power.” Specifically, LaSalle County Station did not have an appropriate analysis to determine the capability of coping with a station blackout in that it had no design basis document that verified the proper operation of the reactor core isolation cooling (RCIC) turbine exhaust pressure trip during station blackout conditions. This issue was entered into the licensee’s corrective action system and the licensee obtained additional information and performed a preliminary analysis which showed that the pressure trip would operate as required.

  This finding was more than minor because it affected an attribute of the mitigating systems cornerstone. Specifically, the licensee had insufficient design control methods in place to demonstrate the operability or reliability of the RCIC turbine exhaust pressure trip during a station blackout. This finding was of very low safety significance because it screened out using the Phase 1 worksheet. (Section 1R21.2.b)

- Green. A finding of very low safety significance was identified by the inspectors associated with a Non-Cited Violation of 10 CFR 50.63, “Loss of All Alternating Current Power.” Specifically, LaSalle County Station had RCIC room station blackout temperature profiles that exceeded the limiting temperature for the skid-mounted RCIC electronic governor module (EGM). This issue was entered into the licensee’s corrective action system and the licensee performed a preliminary analysis which lowered the maximum temperature in the RCIC room. Additionally, the licensee performed testing on the EGM to show that it could operate within the expected temperatures for the required duration.

  This finding was more than minor because it affected an attribute of the mitigating systems cornerstone. Specifically, the licensee had not maintained control of its design such that the capability of the RCIC EGM was invalid. This finding was of very low safety significance because it screened out using the Phase 1 worksheet. (Section 1R21.3.b1)

- Green. A finding of very low safety significance was identified by the inspectors associated with a Non-Cited Violation of 10 CFR 50.63, “Loss of All Alternating Current Power.” Specifically, the licensee did not have an accurate analysis to show that the RCIC pump had sufficient net positive suction head (NPSH) to operate under station blackout conditions. This issue was entered into the licensee’s corrective action system and the licensee performed a preliminary analysis which showed that there was sufficient NPSH during station blackout conditions.

  This finding was more than minor because it affected an attribute of the mitigating systems cornerstone. Specifically, the licensee failed to demonstrate that there was sufficient NPSH available to ensure the operability and reliability of the RCIC pump under station blackout conditions. This finding was of very low safety significance because it screened out using the Phase 1 worksheet. (Section 1R21.3.b2)

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

None.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems and Barrier Integrity

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

.1 Review of 10 CFR 50.59 Evaluations and Screenings

a. Inspection Scope

The inspectors reviewed all three of the full evaluations performed by the licensee pursuant to 10 CFR 50.59 during the last two years. The inspectors reviewed the evaluations to verify that the documents complied with the requirements of 10 CFR 50.59 and that prior NRC approval was not necessary. The inspectors also reviewed 12 screenings where the licensee had determined that a full 10 CFR 50.59 evaluation was not necessary. In regard to these screening evaluations, the inspectors verified that the changes did not constitute an adverse change to an updated final safety analysis report (UFSAR) design function, method of performing or controlling a design function, or an evaluation that demonstrated that an intended function would be accomplished such that a full evaluation under 10 CFR 50.59 would be required in order to determine if prior NRC approval was necessary. The screenings were chosen based on risk significance of samples from the different cornerstones and were not limited to the systems chosen for the safety system design and performance capability (SSDPC) portion of the inspection.

The inspectors used Nuclear Energy Institute (NEI) 96-07, “Guidelines for 10 CFR 50.59 Implementation,” Revision 1, as a reference to determine acceptability of the completed evaluations and screenings. The NEI document was endorsed by the NRC in Regulatory Guide 1.187, “Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments,” dated November 2000. The inspectors also consulted Part 9900 of the NRC Inspection Manual, “10 CFR Guidance for 10 CFR 50.59, Changes, Tests, and Experiments.” The licensee’s “50.59 Resource Manual,” Revision 2, was also consulted during the inspection.

The baseline number of samples could not be completed as the licensee had not performed six to eight full evaluations during the preceding two years. However, this baseline inspection is considered completed as the inspectors reviewed every full 10 CFR 50.59 evaluation performed by the licensee within the revised inspection program (ROP) biennial cycle, the inspectors also reviewed at least the minimum number of screening evaluations, and there was a little likelihood that the licensee would complete additional full evaluations to meet the minimum number of samples by the end of the ROP cycle. The list of documents reviewed by the inspectors is included as an attachment to this report.

b. Findings

Adverse Change to Nitrogen Supply Header UFSAR Description

Introduction: The inspectors identified a Non-Cited Violation (NCV) of 10 CFR 50.59, “Changes, Tests, and Experiments,” which had very low safety significance. Specifically, the licensee failed to complete a full evaluation in accordance with 10 CFR 50.59 for an adverse change to the nitrogen supply header UFSAR description.

Description: The inspectors reviewed screening L04-044 which dealt with revising UFSAR Section 9.3.1.2.2 to clarify the nitrogen supply header function of supporting long-term core cooling via the automatic depressurization system (ADS) valves. The licensee determined that a full evaluation under 10 CFR 50.59 was not necessary.

The UFSAR section 9.3.1.2.2 described the drywell pneumatic system. Revision 14, dated April 2002, stated that the drywell pneumatic system was augmented with two banks of nitrogen bottles that automatically came on line to maintain pressure above 150 pounds per square inch, gauge, (psig) the lower limit for ADS operability. The UFSAR section continued by stating that the purpose of the bottles was to charge the ADS accumulators, enabling ADS operation following an accidental loss of the drywell pneumatic supply. The UFSAR also noted that either bank of bottles had sufficient capability to operate the ADS valves for a week, at the maximum calculated usage. In the February 4, 2004 , approved safety evaluations, the licensee changed this wording to state that the two banks of nitrogen bottles automatically came on line, via pressure regulator valves, to “normally” maintain pressure above 150 psig maintained the header. The purpose of the bottles was changed to “enable ADS long-term operation.” Finally, a new sentence was added which stated that “During pressure regulation control, header pressure may decrease below 150 psig but above 133 psig, which assures that ADS long-term cooling requirements are maintained.”

The inspectors determined that the change appeared to have an adverse impact on the design function of the nitrogen supply header. While the inspectors did not question that the nitrogen supply header had a long term cooling function, the wording of the UFSAR indicated that this was not its only function. Specifically, the UFSAR initially indicated that the purpose of the nitrogen bottles was to charge the accumulators to enable ADS operation following an accidental loss of drywell pneumatic supply. The licensee acknowledged that the accumulators had a short-term safety function. Therefore, the inspectors concluded that the nitrogen supply header also had a short-term safety function as the system automatically came on line to recharge the accumulators anytime the pressure in the accumulators dropped below 150 psig, whether for short or long term applications.

Furthermore, the inspectors noted that the UFSAR words prior to the change indicated that the nitrogen supply header maintained the ADS at or above 150 psig at all times and that the 150 psig was the minimum limit for ADS operability. The inspectors also noted that technical specification surveillance requirement 3.5.1.3 required verification every 31 days that the ADS accumulator supply header pressure was greater than or equal to 150 psig. Although the licensee stated that pressure indicators on the accumulators were used to verify this requirement, the inspectors determined that the proposed change would create a conflict with the technical specification requirement. Specifically, the introduction of the word “normally” implied that there would be times when the supply header pressure could be below 150 psig, such that the surveillance requirement would not be met.

Title 10 CFR 50.59 allows licensees to make changes to the facility providing the change did not require a change to the technical specifications or a license amendment. NEI 96-07 Section 4.2.1 provides guidance on when a full evaluation under 10 CFR 50.59 is necessary. The guidance states, in part, that a full 10 CFR 50.59 evaluation is required for changes that adversely affect design functions. It indicates that changes that have neutral or positive effects may be screened out because only adverse changes have the potential to increase the likelihood of malfunctions, increase consequences, create new accidents or otherwise meet the 10 CFR 50.59 evaluation criteria which might result in a license amendment being required. As the change appeared to have adverse affects, the inspectors concluded that a full 10 CFR 50.59 evaluation should have been completed to evaluate whether a change to the technical specification or a license amendment was required. The failure to have a written evaluation which supported why a license amendment was not needed is a violation of 10 CFR 50.59 Paragraph (d)(1).

Analysis: The team determined that this issue was a performance deficiency since the licensee failed to have a written evaluation as to why a license amendment was not needed, as required under 10 CFR 50.59. The inspectors concluded that the violation was reasonably within the licensee’s ability to foresee and correct because there was readily available guidance which explained how to determine if a change was adverse, and which provided examples illustrating the guidance. The inspectors determined that the licensee had a chance to prevent the violation from occurring as the 10 CFR 50.59 screening received an independent review. Additionally, the licensee performed a self-assessment prior to the inspection which provided an additional opportunity for the licensee to self-identify and correct the deficiency.

Because violations of 10 CFR 50.59 are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process instead of the significance determination process (SDP). However, if possible, the underlying technical issue is evaluated under the SDP to determine the severity of the violation. In this case, the licensee failed to perform a safety evaluation in accordance with 10 CFR 50.59 for adverse changes made to the UFSAR concerning the function of the nitrogen supply header in regard to ADS operation.

The finding was determined to be more than minor because the inspectors could not reasonably determine that the UFSAR change would not have ultimately required NRC approval to a change of technical specification surveillance requirement 3.5.1.3.

The inspectors completed a significance determination of the underlying technical issue using NRC’s inspection manual chapter (IMC) 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations.” The team determined from the mitigating systems evaluation in the Phase 1 screening worksheet that all the questions were answered “No,” therefore the finding was determined to be of very low safety significance (Green). In accordance with the Enforcement Policy, the violation was therefore classified as a Severity Level IV violation.

Enforcement: Title 10 CFR 50.59(d)(1) states, in part, that the licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments. These records must include a written evaluation which provides the bases for the determination that the change, test, or experiment does not require a license amendment. Contrary to the above, on February 4, 2004, the licensee prepared and reviewed a screening for a UFSAR change that introduced adverse changes to the nitrogen supply header design function and failed to perform a safety evaluation in accordance with 10 CFR 50.59. In accordance with the Enforcement Policy, the violation was classified as a Severity Level IV violation because the underlying technical issue was of very low risk significance. Because this non-willful violation was non-repetitive and was captured in the licensee's corrective action program as Issue Report (IR) 353554, this violation is being treated as a Non-Cited Violation consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000373/2005008-01; 05000374/2005008-01)

1R17 Permanent Plant Modifications Biennial Review (71111.17B)

a. Inspection Scope

The inspectors reviewed seven permanent plant modifications which had been installed in the plant during the last two years. The modifications were chosen based primarily upon being within the systems selected for the SSDPC portion of the inspection in order to take dual credit for modification review under both inspection procedures without a corresponding increase in inspection hours. The inspectors reviewed the modifications to verify that the completed design changes were in accordance with the specified design requirements and the licensing bases and to confirm that the changes did not affect any systems' safety function. Design and post-modification testing aspects were verified to ensure the functionality of the modification, its associated system, and any support systems. The inspectors also verified that the modifications performed did not place the plant in an increased risk configuration. The inspectors used applicable industry standards to evaluate acceptability of the modifications. The list of documents reviewed by the inspectors is included as an attachment to this report.

b. Findings

No findings of significance were identified.

1R21 Safety System Design and Performance Capability (71111.21)

Introduction: Inspection of safety system design and performance verifies the initial design and subsequent modifications and provides monitoring of the capability of the selected systems to perform design bases functions. As plants age, the design bases 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 and components selected were the reactor core isolation cooling (RCIC), and the 125/250 volt direct current (DC) distribution systems (two samples). These systems were selected for review based upon:

- having high probabilistic risk analysis rankings;
- considered high safety significant maintenance rule systems; and
- not having received recent NRC review.

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

- licensee technical specifications;
- applicable UFSAR sections; and
- the systems' design documents.

.1 System Requirements

a. Inspection Scope

The inspectors reviewed the UFSAR, technical specifications, system design basis documents, system descriptions, drawings, and other available design basis information, to determine the performance requirements of RCIC and DC distribution systems, and their associated support systems. 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 required review to ensure that the RCIC system was capable of providing adequate core cooling in the event the reactor is isolated from its primary heat sink in conjunction with a loss of normal feedwater flow to the reactor vessel.

Energy Sources: This attribute required review to ensure that the power supply to the RCIC system motor operated valves and other electrical components was adequate for the proper functioning of the valves and other components. This included assuring that the valve power circuit, including the circuit breaker and cable, was adequately sized for the application. For the 125/250 Vdc system this attribute was reviewed to ensure the batteries and the chargers had adequate capacity to support the worst case plant loading. This review also included ensuring that coordination between the load circuit breakers and the feeder breakers to the buses was maintained.

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

Heat Removal: This attribute required review to ensure that the heat generated while the RCIC system was running can be effectively removed and that the temperature in the battery rooms would be maintained within the batteries' design requirements.

b. Findings

b.1 Battery Charger Loading Configuration

Introduction: The inspectors identified an NCV having very low safety significance (Green) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." Specifically, LaSalle County Station failed to maintain an accurate design basis supporting the addition of loads due to the simultaneous energization of both battery chargers. The licensee performed an informal calculation which showed that the diesel loading would have been acceptable even with both battery chargers energized.

Description: The inspectors identified that, during battery charger troubleshooting, the licensee operated in a plant configuration that was not supported by the licensee's design basis documentation. Specifically, the licensee operated with both the main and backup battery chargers simultaneously loaded onto the associated alternating current (AC) buses. The licensee determined this to be appropriate because the period for troubleshooting was to be less than three hours in duration. Therefore, the battery chargers were regarded as intermittent loads and did not have to be considered from an electrical loading perspective. The licensee documented their rationale in engineering change request (ECR) 368281; however, the licensee did not consider an ECR as design basis documentation and the ECR did not contain sufficient information to justify the acceptability of the intermittent load concept.

The inspectors were concerned with the justification and conclusions drawn from this ECR, because the additional loads, although only present for a short duration, could potentially affect voltage drop during the initial stages of a loss of coolant event (LOCA) event with off-site power available. No formal documented calculation or evaluation existed to ensure that safety-related loads would function properly during a LOCA with these additional loads on line. Additionally, the inclusion of the extra charger was a fairly significant load addition, adding a maximum additional loading of 28.3 kilowatts. The inspectors noted that the duration of the extra loads was not appropriate justification if the addition caused equipment to be inoperable. Additionally, while this ECR was written only for the battery chargers, the inspectors recognized that the logic used, as written in ECR 368281, could be used to support any number of other, potentially larger, loads being energized at one time, as long as those loads were only connected for a short duration.

Based upon the inspectors concerns, the licensee issued corrective action document IR 353537. Also, the licensee performed an engineering justification (EC 356294) to support operation with the two battery chargers energized that considered loading and included voltage drop considerations during LOCA block starts. While the licensee eventually was able to conclude that operating with both chargers energized was acceptable, there was no documented basis for this configuration prior to the inspection.

Analysis: The inspectors determined that this issue was a performance deficiency since the licensee failed to meet the requirements of 10 CFR Part 50 Appendix B, Criterion III. Specifically, the licensee did not maintain an accurate design basis to show that AC system loading following a design basis accident could handle the simultaneous energization of both battery chargers. The cause was reasonably within the licensee's ability to foresee and correct and it could have been prevented because the licensee had procedures which addressed adding intermittent loads to safety-related buses. Additionally, the licensee performed a self-assessment prior to the inspection which provided an additional opportunity for the licensee to self-identify and correct the deficiency.

The issue was more than minor because it was associated with and affected the mitigating systems cornerstone. Specifically, the licensee could not initially demonstrate that the design basis of the plant was not affected by adding the additional battery charger load. The finding screened as having very low significance (Green) using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for the At-Power Situations," because the inspectors answered "no" to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," states, in part, that measures shall be established to assure that applicable design basis are correctly translated into specifications, drawings, procedures and, instructions. It also states that this measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from those standards are controlled. The design basis of the LaSalle County Station described the AC system loading prior to an accident as only having one battery charger per division energized to ensure that the AC system could handle accident loads. This design standard was properly translated into station procedures and corporate procedures provided direction on deviating from that standard.

Contrary to the above, in January 2005, the licensee simultaneously energized two battery chargers on the same division and failed to control a deviation in the allowed electrical loading. Specifically, the licensee only considered the time the additional load was energized and not the effect on the accident loading. Because the failure to control the deviation from the design standard was determined to be of very low safety significance and because it was entered in the licensee's corrective action program as IR 353537, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000373/2005008-02; 05000374/2005008-02).

b.2 Water Leg Pump Room Heatup Calculation Non-Conservatisms

Introduction: The inspectors identified an NCV having very low safety significance (Green) of 10 CFR 50.63, "Loss of All Alternating Current Power." Specifically, LaSalle County Station failed to maintain an accurate design basis heat-up calculation that supported the heat loads that would be present during a station blackout event for the water leg pump room.

Description: The inspectors identified that the licensee did not have a calculation to support the heatup rate experienced during a station blackout event in the water leg pump room. This calculation was required in order to accurately determine the environmental effects on the operability and reliability of equipment necessary to cope with the station blackout. Although the licensee had calculations that purported to address the heatup rate in the water leg room, the inspectors identified that none of these heatup rate calculations accounted for all of the heat sources that would be encountered during a station blackout. Specifically, the existing calculations did not account for both the electrical heat loads in the room and the steam that would be emitted into the room from the RCIC room located below.

In response to this concern, the licensee issued corrective action document IR 354050 to document and correct this issue. The licensee was able to account for the non-conservatisms that the inspectors identified by using conservatisms already present in the heatup calculations of record. However, the inspectors noted that the licensee also used a more modern “realistic” computer code to calculate the maximum temperatures in place of the original computer code. Between the use of the modern code and removal of conservatisms, the licensee was able to conclude that the already established temperature of 229.6 degrees Fahrenheit (EF) for the RCIC water leg room was still a bounding value for the station blackout event. Based upon this information, the inspectors determined that the licensee’s established environmental evaluations would still apply since the bounding temperature for the room had not changed.

Analysis: The team determined that this issue was a performance deficiency since the licensee failed to meet the requirements of having an appropriate analysis to determine their capability of coping with a station blackout. Specifically, the licensee did not have an accurate design basis heat-up calculation that supported the actual heat loads that would be present during a station blackout event in the water leg pump room. The cause was reasonably within the licensee’s ability to foresee and correct and it could have been prevented because the licensee had an opportunity to review the calculation during the request for and approval of the power uprate license amendment. Additionally, the licensee performed a self-assessment prior to the inspection which provided an additional opportunity for the licensee to self-identify and correct the deficiency.

The finding is more than minor because it is associated with and affects the mitigating systems cornerstone. Specifically, the licensee had not maintained design control over the maximum heatup temperature in the water leg pump room which are necessary for coping with a station blackout. This required the licensee to perform a new calculation which took into account the additional heat loads and steam leak into the room. This finding screened as of very low significant (Green) using IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for the At-Power Situations,” because the inspectors answered “no” to all five questions under the Mitigating Systems Cornerstone column.

Enforcement: Title 10 CFR 50.63, "Loss of All Alternating Current Power," Paragraph (a)(2) requires, in part, that licensees provide sufficient capacity and capability to ensure the core is cooled in the event of a station blackout for the specified duration. It further requires that the capability for coping with a station blackout of specified duration shall be determined by an appropriate coping analysis. Finally, it requires that licensees have the baseline assumptions, analyses, and related information used in their coping evaluations available for NRC review.

Contrary to the above, as of July 14, 2005, the licensee did not have an appropriate coping analysis which determined the capability of components in the RCIC water leg pump room to operate during a station blackout. Specifically, the licensee failed to have an analysis which accounted for all heat loads into the water leg pump room. Because this violation is of very low safety significance and because LaSalle Station has entered this finding into its corrective action program (IR 354050), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000373/2005008-03; 05000374/2005008-03).

.2 System Condition and Capability

a. Inspection Scope

The inspectors reviewed design basis documents and plant drawings, abnormal and emergency operating procedures, requirements, and commitments identified in the UFSAR and technical specifications. The inspectors compared the information in these documents to applicable electrical, instrumentation and control, mechanical calculations, setpoint changes, and plant modifications. The inspectors used applicable industry standards, such as the American Society of Mechanical Engineers (ASME) Code and the Institute of Electrical and Electronics Engineers (IEEE), to evaluate acceptability of the systems' design. Select operating experience was reviewed to ensure the issue was adequately evaluated and corrective actions 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 the actual system condition and tested capability were consistent with the identified design bases. Specifically, the inspectors reviewed the installed configuration, the system operation, the detailed design, and the system testing, as described below.

Installed Configuration: The inspectors confirmed that the installed configuration of the RCIC and DC power systems met the design basis by performing detailed system walkdowns. 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; battery physical separation; provisions for seismic stability of the batteries; likelihood of pressure transients on RCIC; and the conformance of the currently installed configuration of the systems with the design and licensing bases. The walkdowns also verified instrument settings and the appropriateness of design input values.

Operation: The inspectors verified that the RCIC and DC systems were operated in accordance with design basis documents and station procedures. The inspectors evaluated the effects on the system of temporary changes or equipment being out of service and ensured that operations staff would have required access to equipment if needed during postulated scenarios.

Design: The inspectors reviewed the mechanical, electrical, and instrumentation design of the RCIC and DC power distribution systems to verify that the systems and subsystems would function as required under design conditions. This 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 setpoints 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 that testing was consistent with design basis information.

b. Findings

RCIC Exhaust Pressure Trip Instrumentation

Introduction: The inspectors identified an NCV having very low safety significance (Green) of 10 CFR 50.63, "Loss of All Alternating Current Power." Specifically, LaSalle County Station did not have an appropriate analysis to determine the capability of coping with a station blackout in that it had no design basis document that verified the proper operation of the RCIC turbine exhaust pressure trip during these conditions.

Description: While reviewing the setpoint calculation for the Units 1 and 2 RCIC turbine exhaust high pressure turbine trip, the inspectors questioned whether the instrument would perform properly at the elevated temperatures that the instrumentation would experience during a station blackout. During their review, the inspectors observed that the "Design Inputs" section of the setpoint calculation included the following statement: "In this calculation, temperature error will be evaluated up to 145EF. If the temperature ever exceeds 145EF, this calculation is not valid." Since the licensee had already determined, by calculation, that the temperature in the RCIC room where the instrumentation was located could be as high as 206.4EF during a station blackout, the inspectors were concerned that the licensee had no design basis document that verified the proper operation of the RCIC turbine exhaust pressure trip. Specifically, the inspectors were concerned that the instrumentation could behave so erratically at these heightened temperatures that the RCIC turbine could trip during a station blackout leaving the plant without its primary means of maintaining reactor coolant inventory.

Based upon the inspectors' concerns, the licensee further researched the issue and discovered that there was no established basis for the operation of this instrumentation above 145EF. Because of this design deficiency, the licensee initiated corrective action document IR 351884. This document performed an evaluation for the basis of operability for the instrumentation. For that evaluation, the licensee obtained test data from external sources that verified operation of identical models of pressure switches in temperatures as great as 212EF. Based upon this test information, the licensee concluded that the RCIC turbine exhaust pressure trip would operate properly for the heightened temperatures achieved during station blackout conditions.

While the licensee was able to determine operability of the exhaust pressure trip by obtaining test data from external sources, the licensee's existing design basis had not been adequate. Prior to the inspectors' questioning the operation of the trip instrumentation above 145EF, the licensee did not have a basis that supported the proper operation of this trip function during a station blackout event.

<u>Analysis</u>: The team determined that this issue was a performance deficiency since the licensee failed to meet the requirements of having an appropriate analysis to determine their capability of coping with a station blackout. Specifically, the licensee did not have a design basis document that verified the proper operation of the RCIC turbine exhaust pressure trip during these conditions. The cause was reasonably within the licensee's ability to foresee and correct and it could have been prevented because the licensee had an opportunity to identify the issue when it raised the pressure trip setpoint in 1993 and again in 2000 when reviewing calculations for the power uprate amendment. Additionally, the licensee performed a self-assessment prior to the inspection which provided an additional opportunity for the licensee to self-identify and correct the deficiency.

The issue was determined to be more than minor because it was associated with and affected the mitigating systems cornerstone. Specifically, the licensee had insufficient design control methods in place to demonstrate the operability or reliability of the RCIC turbine exhaust pressure trip during a station blackout. Failure of the turbine exhaust pressure switches due to the temperatures being above the analyzed limit for the switch could have tripped the RCIC pump. The licensee had to obtain additional data and perform an operability evaluation for the instrumentation. This finding screened as of very low significant (Green) using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for the At-Power Situations," because the inspectors answered "no" to all five questions under the Mitigating Systems Cornerstone column in the Phase 1 worksheet.

<u>Enforcement</u>: Title 10 CFR 50.63, "Loss of All Alternating Current Power," Paragraph (a)(2) requires, in part, that licensees provide sufficient capacity and capability to ensure the core is cooled in the event of a station blackout for the specified duration. It further requires that the capability for coping with a station blackout of specified duration shall be determined by an appropriate coping analysis. Finally it requires that licensees have the baseline assumptions, analyses, and related information used in their coping evaluations available for NRC review.

Contrary to the above, as of July 15, 2005, the licensee failed to have an appropriate coping analysis which determined the capability of the RCIC pump to operate during a station blackout. Specifically, the licensee failed to have an analysis which verified that the appropriate operation of the RCIC turbine exhaust pressure trip during a station blackout. Because the violation was of very low safety significance and because the licensee entered the finding into their corrective action system as IR 351884, this violation is being treated as a Non-Cited Violation, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000373/2005008-04; 05000374/2005008-04).

.3 Components

a. Inspection Scope

The inspectors examined the RCIC and the DC power distribution systems to ensure that component level attributes were satisfied. The inspectors specifically focused on the batteries and battery chargers in the DC system, and on the RCIC pump and turbine in the RCIC system. The following component level attributes of the RCIC and DC power distribution systems were reviewed:

Component Degradation: This attribute was reviewed to ensure that components were being maintained consistent with the design basis. The inspectors reviewed RCIC and DC battery surveillance tests to ensure that equipment degradation, if present, was within allowable limits. The inspectors also verified that component replacement was within its expected life and that no components were not being replaced at an excessive frequency indicative of underlying problems.

Component Inputs/Outputs: The inspectors reviewed component specific inputs and outputs to verify that the components would operate acceptably under accident conditions.

Equipment/Environmental Qualification: This attribute verifies that the equipment is qualified to operate under the environment in which it is expected to be subjected to under normal and accident conditions. The inspectors reviewed design information, specifications, and other documentation to ensure that the RCIC and the DC power distribution components were qualified to operate within the temperatures specified in the station blackout documentation.

Equipment Protection: This attribute verifies that the RCIC and the DC power distribution systems are adequately protected from natural phenomenon and other hazards, such as high energy line breaks, floods or missiles. The inspectors reviewed design information, specifications, and documentation to ensure that the RCIC and the DC power distribution systems were adequately protected from those hazards identified in the UFSAR which could impact their ability to perform their safety function.

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 and performed physical walkdowns to ensure any operating experience described conditions either did not exist or had been identified and corrected.

b. Findings

b.1 RCIC Electronic Governor Modules

Introduction: The inspectors identified an NCV having very low safety significance (Green) of 10 CFR 50.63, “Loss of All Alternating Current Power.” Specifically, LaSalle County Station had RCIC room station blackout temperature profiles that exceeded the limiting temperature for the skid-mounted RCIC electronic governor module (EGM).

Description: During review of the environmental conditions for the RCIC pump and turbine, the inspectors determined that the licensee did not have documentation to show that the EGM would be able to operate at the RCIC room temperature for a station blackout event. The EGM was a skid-mounted module that provided control signals for the RCIC woodward governor system. Failure of the EGM would result in a loss of speed control for the RCIC turbine. This would result in an overspeed and mechanical overspeed trip.

The inspectors ascertained that the maximum expected room temperature during a station blackout was 206.4EF as determined in the RCIC room heat up calculation, ATD-351, “RCIC Pump Room Temperature Transient Following Station Blackout with Gland Seal Leakage.” The inspectors also determined that the licensee had claimed that all equipment in the RCIC room was qualified to 212EF as part of the power uprate license amendment submittal. However, the licensee was not able to produce any documentation which supported qualification of the EGM past 150EF.

In response to this identified issue, the licensee wrote operability evaluation 05-006, “RCIC Operability Determination.” In addition, Calculation ATD-351 was revised to develop a more realistic room temperature response. The results of the revised analysis concluded that the maximum RCIC room temperature during the station blackout coping period of 4 hours and 15 minutes would not exceed 165EF. As this was still above the 150EF qualification temperature for the EGM electronic module, the licensee conducted an elevated temperature operability test, SEAG 05-000069, “LaSalle Special Test of RCIC Electronic Controls EGM/RGSC,” on July 15, 2005. This test simulated RCIC operation during a station blackout event in a controlled temperature environment using a spare RCIC EGM. The licensee demonstrated that the RCIC EGM would be able to perform its intended function during a station blackout for temperatures up to 169EF for the duration of the station blackout. The licensee documented the acceptability of the EGM during a station blackout in EC 356324, “Evaluation of the Capability of Reactor Core Isolation Cooling Governor EGM/RGSC to Operate in Station Blackout Environment.”

Analysis: The team determined that this issue was a performance deficiency since the licensee did not have an appropriate coping evaluation to show that equipment required for a station blackout would be available when required. Specifically, the existing coping analysis was not appropriate as it did not show that the RCIC turbine would operate for the required 4 hours and 15 minutes at the room temperature postulated during a station blackout. The cause was reasonably within the licensee’s ability to foresee and correct and it could have been prevented because the licensee had an opportunity to identify the issue when the licensee submitted a license amendment for power uprate and specifically reviewed the environmental conditions of the equipment in the RCIC room. Additionally, the licensee performed a self-assessment prior to the inspection which provided an additional opportunity for the licensee to self-identify and correct the deficiency.

The finding was more than minor because it was associated with the mitigating system cornerstone attributes of design control and equipment performance and affected the objective of ensuring the capability of the RCIC system in performing its design basis function. Specifically, the licensee had not maintained control of its design such that the capability of the RCIC EGM was not demonstrated until the inspectors questioned it. This finding screened as having very low significant (Green) using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for the At-Power Situations," because the inspectors answered "no" to all five questions under the Mitigating Systems Cornerstone column.

Enforcement: Title 10 CFR 50.63, "Loss of All Alternating Current Power," Paragraph (a)(2) requires, in part, that licensees provide sufficient capacity and capability to ensure the core is cooled in the event of a station blackout for the specified duration. It further requires that the capability for coping with a station blackout of specified duration shall be determined by an appropriate coping analysis. Finally it requires that licensees have the baseline assumptions, analyses, and related information used in their coping evaluations available for NRC review.

Contrary to the above, as of July 14, 2005, the licensee did not have an appropriate coping analysis which determined the capability of the RCIC turbine to operate during a station blackout. Specifically, the licensee failed to have documentation supporting the baseline assumption that all equipment in the RCIC room was qualified for the environment under which it had to operate during a station blackout. Because this violation is of very low safety significance and because LaSalle County Station has entered this finding into its corrective action program (IR 353163), this violation is being treated as a Non-Cited Violation, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000373/2005008-05; 05000374/2005008-05).

b.2 Net Positive Suction Head (NPSH) of the RCIC Pump

Introduction: The inspectors identified an NCV having very low safety significance (Green) of 10 CFR 50.63, "Loss of All Alternating Current Power." Specifically, the licensee did not have an accurate analysis to show that the RCIC pump had sufficient net positive suction head (NPSH) to operate under station blackout conditions.

Description: The inspectors identified that the licensee failed to use the correct pump operating curve to determine the required NPSH for the RCIC pump. The licensee used an operating curve that was developed for pumps operating at constant revolutions per minute (RPM) significantly lower than the RPM where the RCIC pumps were required to operate. Use of the incorrect operating curve resulted in a lower required NPSH being specified as an acceptance criteria than the actual required NPSH for the operating pumps. Specifically, in Calculation L-002540, the licensee determined the available NPSH under station blackout conditions; the calculated available NPSH was then compared to the incorrect required NPSH to show that the pumps would operate satisfactorily under accident conditions.

The licensee contacted the manufacturer to obtain the required NPSH value at the operating RPM. The licensee compared this value with the calculated available NPSH for a station blackout and determined that, per the calculation, there was insufficient available NPSH to match the actual required NPSH. The licensee issued an operability evaluation OE 05-005 to determine the operability of the RCIC pump during a station blackout. The operability evaluation recalculated the available NPSH, eliminating some of the conservatisms, most particularly raising the minimum level in the suppression pool following a station blackout event. This allowed the licensee to conclude that the RCIC pump was still operable during a station blackout given the actual operating conditions.

While the licensee was ultimately able to show that the calculated available NPSH for the RCIC pump during a station blackout was still sufficient to ensure pump operability, the licensee’s existing design basis had not been adequate. In order to meet the true acceptance criteria for the required NPSH, the licensee had to first obtain the correct NPSH required and then to redo the available NPSH calculation and remove conservatisms in order to show that sufficient NPSH would actually be available under station blackout conditions.

Analysis: The team determined that this issue was a performance deficiency since the licensee failed to meet the requirements of having an appropriate analysis to determine the capability of the RCIC pumps to operated during a station blackout. Specifically, the licensee did not have an accurate calculation to determine the available NPSH for the RCIC pump under station blackout conditions. The cause was reasonably within the licensee’s ability to foresee and correct and it could have been prevented because the operating curve for the RCIC pump clearly stated the RPM for which the values in the curve were applicable for that specific RMP and the licensee had revised the calculation following the power uprate license amendment and lowered the required NPSH during the revision. Additionally, the licensee performed a self-assessment prior to the inspection which provided an additional opportunity for the licensee to self-identify and correct the deficiency.

The issue is more than minor because it is associated with and affects the mitigating systems cornerstone objective of design control. Specifically, the licensee could not demonstrate that there was sufficient NPSH available to ensure the operability and reliability of the RCIC pump under station blackout conditions. This required the licensee to reperform the calculation with the correct values, removing conservatisms to ensure sufficient available NPSH existed. This finding screened as having very low significance (Green) using IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for the At-Power Situations,” because the inspectors answered “no” to all five questions under the Mitigating Systems Cornerstone column.

Enforcement: Title 10 CFR 50.63, “Loss of All Alternating Current Power,” Paragraph (a)(2) requires, in part, that licensees provide sufficient capacity and capability to ensure the core is cooled in the event of a station blackout for the specified duration. It further requires that the capability for coping with a station blackout of specified duration shall be determined by an appropriate coping analysis. Finally it requires that licensees have the baseline assumptions, analyses, and related information used in their coping evaluations available for NRC review.

Contrary to the above, as of July 15, 2005, the licensee failed to have a coping analysis which demonstrated the capability of the RCIC pump to operate during a station blackout. Specifically, the analysis failed to demonstrate that there was sufficient NPSH available to ensure the operability of the RCIC pump under station blackout conditions. Because the violation was of very low safety significance and because the licensee entered the finding into their corrective action system as IR 352743, this violation is being treated as a Non-Cited Violation, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000373/2005008-06; 05000374/2005008-06).

4. OTHER ACTIVITIES (OA)

4OA2 Problem Identification and Resolution

Review of Condition Reports

a. Inspection Scope

The inspectors reviewed a sample of RCIC and DC power distribution system problems 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, condition 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.

4OA6 Meetings, Including Exits

.1 Exit Meeting

The inspectors presented the inspection results to Ms. Susan Landahl and other members of licensee management at the conclusion of the inspection on July 26, 2005. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

S. Landahl, Site Vice President
D. Enright, Plant Manager
R. Chrzanowski, Chemistry Manager
L. Coyle, Operations Director
D. Czufin, Site Engineering Director
B. Ginter, Electrical Design Engineering Manager
F. Gogliotti, System Engineering Manager
P. Holland, Regulatory Compliance
B. Hilton, Mechanical & Structural Design Engineering Manager
M. Murskyi, Electrical System Engineering Manager
J. Rommel, Mechanical Design Engineering Manager
T. Simpkin, Regulatory Compliance Manager

Nuclear Regulatory Commission
C. Pederson, Director, Division of Reactor Safety
A. M. Stone, Chief, Engineering Branch 2, Division of Reactor Safety
D. Kimble, Senior Resident Inspector
D. Eskins, Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000373/2005008-01; 05000374/2005008-01	NCV	Failure to Perform 10 CFR 50.59 Evaluation for an Adverse Change to the UFSAR
05000373/2005008-02; 05000374/2005008-02	NCV	Inadequate Design Basis for Simultaneous Energization of Both Battery Chargers
05000373/2005008-03; 05000374/2005008-03	NCV	Inadequate Water Leg Pump Room Heatup Calculation
05000373/2005008-04; 05000374/2005008-04	NCV	Inadequate Setpoint Calculation Associated with the RCIC Turbine Exhaust Pressure Trip
05000373/2005008-05; 05000374/2005008-05	NCV	Inadequate Temperature Qualifications for RCIC Electronic Governor Modules
05000373/2005008-06; 05000374/2005008-06	NCV	Inadequate NPSH for the RCIC Pump

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection, including 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.

1R02 Changes, Tests, and Experiments

Drawings

115D6404; Drive Mechanism Elementary Diagram; Revision 5

1E-1-4213AC; Traversing In-core Probe Calibration System "NR" (C51D) Schematic Diagram Part 3; Revision D

M-98; Fuel Pool Cooling Filter and Demineralizing System Piping and Instrumentation Diagram, Sheet 4; Revision N

M-99; Standby Liquid Control System Piping and Instrumentation Diagram; Revision AA

M-100; Control Rod Drive Hydraulic Piping and Instrumentation Diagram; Revision AA

M-144; Fuel Pool Cooling Filter and Demineralizing System Piping and Instrumentation Diagram, Sheet 1; Revision AJ; and Sheet 2; Revision V

M-841; Fuel Pool Cooling Filtering and Demineralizing Piping; Revision J

M-863; Reactor Building Miscellaneous Piping – Miscellaneous Plans and Sections, Sheet 54; Revision Y

M-1163; Reactor Building Miscellaneous Piping Support Location – Miscellaneous Plans; Revision Y

SC-1; Spent Fuel Pool Leak Off Line Isometric Diagram; Revision F

S-783; Reactor Building Pool Liner Sections and Details, Sheet 1; Revision H

S-784; Reactor Building Pool Liner Sections and Details, Sheet 2; Revision L

Engineering Changes

332208; Evaluation of Appendix J Testing Requirements on the Standby Liquid Control System; dated March 31, 2005

343798; Evaluation of Scram Discharge Volume Hydrolazing; dated August 21, 2003

346751[1]; Thermocouple Cut at Reactor Pressure Vessel; dated May 31, 2005

Issue Reports Written as a Result of the Inspection

353545; Hydrolazing Procedure Issue; dated July 15, 2005

353548; Removal of Standby Liquid Control from Type C Testing; dated July 15, 2005

353554; UFSAR Change to the IN System Was Adverse Change Under 10 CFR 50.59; dated July 15, 2005

Issue Reports Reviewed During the Inspection

254171; Cracks in DuraLife 215 Control Rod Blades; dated September 22, 2004

Miscellaneous

AT 184507-04; Evaluate Acceptability of Unit 2 Fuel Pool Liner Leak; dated June 30, 2004

LUCR-34; UFSAR Change Request for Spent Fuel Pool Leakage Description; dated September 16, 2004

Regulatory Review of LaSalle Licensing Basis for Appendix J Testing: White Paper; undated

Type C Testing Results for Standby Liquid Control Containment Isolation Valves; dated April 18, 2005

Operability Evaluations

04-003; Nitrogen Bottle Bank Pressure Regulator; dated May 21, 2004

04-010; Evaluate Cracks Found in DuraLife 215 Control Rod Blades; dated October 6, 2004

Procedures

LMP-RD-08; Hydrolazing Scram Discharge Volume Header; dated September 23, 2004

LOA-RM-101; Unit 1 Rod Manual Control System Abnormal Situations; dated August 30, 2004

LOP-NB-02; Operations with the Potential to Drain the Reactor Vessel; dated February 8, 2005

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[1] This engineering change was also credited as part of the review for baseline procedure 71111.17B.

OU-AA-103; Shutdown Safety Management Program; Revision 4

WC-AA-106; Attachment 4, Job Types; Revision 2

10 CFR 50.59 Evaluations

M-1-0-09-021[2]; Demolition of Hypochlorite System and Installation of New Chemical Feed System; dated November 20, 1990

L01-0367[2]; Install Vendor Supplied Chemical Feed Pump House; dated May 16, 2001

L03-0335; Permit Hydrolazing of the Scram Discharge Volume Headers Without Closing the 1(2)C11-DXXXYY-112 Manual Isolation Valves; dated October 8, 2003

L04-014; LaSalle Unit 1 Cycle 11 Reload Package; dated January 15, 2004

L05-073; LaSalle Unit 2 Cycle 11 Reload Package; dated March 3, 2005

10 CFR 50.59 Screenings

L04-011; Replace Main Steam Line Drain Line Orifices; dated January 13, 2004

L04-044; Revise UFSAR to Reflect Lower Limit for ADS Nitrogen Supply Header Pressure; dated February 4, 2004

L04-053; Cracking of DuraLife 215 Control Blades; dated February 14, 2005

L04-174; Revise Procedure Attachment to Accommodate Alternate Determination of Full in Control Rod Indication on Multiple Rods; dated July 21, 2004

L04-208; Revise Wording in UFSAR Section 9.1.2.1.3 to Be Consistent with that in UFSAR Appendix B; dated September 10, 2004

L04-216; Traversing In-core Probe System Proximity Switch and Channel Selector Upgrades; dated September 20, 2004

L04-226; Capacity and Operating Plan for Cooling Lake; dated September 30, 2004

L04-272; Incorporation of GE14 Fuel Storage Criteria into the LaSalle Licensing Basis for the Unit 2 Spent Fuel Pool; dated November 10, 2004

L05-038; Revise Unit 2 Power to Flow Map; dated February 2, 2005

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[2] This evaluation was not credited as part of the review for baseline procedure 71111.02 as it was completed prior to the revised oversight program biennial cycle under review.

L05-055; Revise Reactor Power Versus Allowed Steam Dome Pressure Map in Procedure LOA-EH-101; dated February 15, 2005

L05-062; Elimination of Type C Testing of Type C Testing on Standby Liquid Control Containment Isolation Valves; dated February 18, 2005

L05-074; LaSalle Units 1 and 2 GE 14 Fuel Implementation; dated February 28, 2005

1R17 Permanent Plant Modifications

Engineering Changes[3]

332600; Replace the 2E51-F052 and 2E51-F053 Double Block Valves; dated March 17, 2005

333812; Install Back-Up Unit 1 Division 2 125 Vdc Battery Charger; Revision 0

339402[4]; Remove Position Indication from 1E51-F066 and Move Associated Main Turbine Trip Logic to 1E51-F065; dated February 9, 2004

339471[4]; Provide Vent Line from RCIC Injection to RX Vessel Vent Line GE SIL 643 Hydrogen Combustion; dated February 24, 2003

341074; Remove Valve 2E51-F357 from Piping Line 2RI09A-2" and Install Blind Flanges and Pipe in its Place; Revision 0

346133[4]; Replace RCIC Orifice 1E51-D307; Revision 2

346751; Thermocouple Cut at Reactor Pressure Vessel; dated May 31, 2005

1R21 Safety System Design and Performance Capability

Calculations

3C7-0189-001; Station Blackout Condensate Inventory Coping Assessment; Revision 3

3C7-0283-001; Extended Blowdown Test Evaluation of Suppression Pool Temperature Measurements; Revision 0

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[3] The licensee uses the term “engineering changes” to describe various documents, not all of which involve modifications to the plant. Only the engineering changes specified in this section were deemed by the team to involve permanent changes to the plant and were thus credited towards completing the baseline program review.

[4] These modifications were also credited as part of the review for baseline procedure 71111.21.

3C7-0390-002; Drywell Temperature Following a Station Blackout; Revision 3

4266/19AI27; Electrical Heat Loads During Station Blackout for Areas Adjacent to Main Control Room, RCIC Room and Auxiliary Electrical Equipment Room; Revision 2

4266/19AI29; Effect of Elevated Temperatures During Station Blackout on Safety Relief Valves and RCIC Pumps/Seals; Revision 0

4266/19AZ31; Degraded Voltage with a 2.5 Percent Boost at the Division 1/2 Unit Substation Transformer; Revision 0

4266/19D30; Capability of 125V and 250V Batteries to Feed Loads During Station Blackout; Revision 3

4266/19D51; Evaluation of the Replacement RCIC Barometric Condenser Condensate Motor; Revision 0

4266/19G-5; Power Cable Ampacities, 8 kV, 5 kV, 600V; Revision 1

4266/AD30; Capability of 125V and 250V Batteries to Feed Loads During Station Blackout; Revision 0

88-088/88-091; Minimum RCIC Pump Recirculation; Revision 0

91-0044; Available NPSH in RCIC System; Revision 0

ATD-0351; RCIC Pump Room Temperature Transient Following Station Blackout With Gland Seal Leakage; Revision 1

AZ48; Summary for 480 V Loads Method of Resolution; Revision A

CID-RI-01; RCIC Pump Discharge Flow Control Error; Revision 0

CQD-055096; Calculation for Environmental Qualification Impact Due to Station Blackout Conditions; Revision 00

D11; DC Distribution Equipment Breaker and Motor Control Center Settings; Revision 43B

D24; Battery and Charger Discharge Rate Alarm Setpoints; Revision 4

D27; 125V Division 1 Battery Sizing; Revision 012B

D33; 250 VDC System Short Circuit; Revision 1

D34; 125 VDC System Short Circuit; Revision 2

D4; Sizing Battery Chargers for 125 V GNB NCX-17 Batteries 2A, 2B; Revision I02

L-000041; Motor Operated Valve Motor Terminal Voltage Calculation for RCIC Valves; Revision 1B

L-000200; Motor Operated Valve Motor Terminal Voltage Calculations for E51 (RCIC) System; Revision 0

L-000488; Vortex Limit and RCIC Suction Transfer Setpoint For Cycled Condensate Storage Tank; Revision 0

L-001024; Low Pressure Core Spray Pump Cubicle Cooler Ventilation System; Revision 2

L-001947; Safe Shutdown Control Circuit Breaker-Fuse Coordination; Revision 2

L-002394; Effect of Higher Terminal DC Voltage on Qualified Life of Environmentally Qualified Equipment; Revision 00

L-002466; Instrument Drift Analysis for Robertshaw Model SP-222-C Pressure Switches; Revision 0

L-002489; Suppression Pool Temperature Transient Analysis; Revision 3

L-002540; NPSH Margin for High Pressure Core Spray, Residual Heat Removal, and RCIC Pumps, Backpressure for RCIC Turbine; Revision 0

L-002590; Condensate Storage Tank Level Switch Setpoint Error Analysis; Revision1

L-002593; Instrument Setpoint Analysis for RCIC Steam Flow High Time Delay; Revision 1

L-002968; DC System Ground Detector Action Levels; Revision 0

L-002996; RCIC DC Motor Methodology; Revision 1

MES-7.2; Piping Heat Losses - Insulated and Uninsulated; Revision B

NED-E-EIC-0084; Units 1 and 2, Division 1 125 Vdc Battery Charger Low and High Voltage Relay Settings; Revision 0

RI-08; RCIC Pump Available NPSH; Revision 1

RI-16; Orifice Sizing of 1E51-D006 and 1E51-D307; Revision 1A

VX-09; Battery Rooms Hydrogen Concentration; Revision 12

<u>Drawings</u>

21A9243BU; Reactor Core Isolation Cooling Pump (RCIC) - Data Sheet; Revision 4

1E-1-4224 Series; Schematic Diagram Leak Detection System “LD” (E31); Various

1E-1-4226 Series; Schematic Diagram Reactor Core Isolation Cooling System “RI” (E51); Various

1E-2-4000DB; Station Key Diagram 125 Volt DC Distribution System; Revision H

1E-2-4000FC; Key Diagram 125 V DC Distribution- ESS Div. 2; Revision N

1E-2-4008AK; Schematic Diagram Div. 2 125V DC Battery Main Charger (2BA) (2DC17E) System “DC” Part 3; Revision C

1E-2-4008ZC; Loop Schematic Diagram 125 V DC Battery (Division 2) System “DC”; Revision E

1E-2-4226AQ; Schematic Diagram Reactor Core Isolation Cooling System “RI” (E51) Part 15; Revision R

1E-2-4226AU; Schematic Diagram Reactor Core Isolation Cooling System “RI” (E51) Part 19; Revision W

1E-2-4226AV; Schematic Diagram Reactor Core Isolation Cooling System “RI” (E51) Part 20; Revision N

22A2869AF; Reactor Core Isolation Cooling System Design Specification Data Sheet; Revision 15

31066; Characteristic Curve Sheet Bingham Pump Division; dated August 23, 1972

A-219; Reactor Building Basement Floor Plan East Area; Revision M

A-223; Reactor Building Upper Basement Floor Plan East Area; Revision N

A-227; Reactor Building Ground Floor Plan East Area; Revision U

FD 210013; Unit Data Pump Specification; dated February 25, 1971

GEK-63042B; Figure 2-4: RCIC Auto and Manual Isolation Logic Diagram

M-74; Cycled Condensate Storage; Revision AM

M-93; Nuclear Boiler and Reactor Recirculating System; Revision AV

M-101; Reactor Core Isolation Coolant System; Revision AN

M-766; Outdoor Piping; Revision W

Engineering Changes

EC 335449; Evaluation of LOS-RI-R3 Specified Performance Requirements; dated February 25, 2002

EC 339402; Remove Position Indication From 1E51-F066 and Move the Associated Main Turbine Trip Logic to 1E51-F065; dated February 9, 2004

EC 339471; Provide Vent Line From RCIC Injection to Reactor Vessel Vent Line - GE Sil 643 Hydrogen Combustion; dated February 17, 2003

EC 340584; Evaluate Acceptability of Energizing Both 125 VDC Division 1 or 2 Battery Chargers Simultaneously to Support Battery Chargers Testing; Revision 0

EC 344824; RCIC Suction Piping Evaluation for Pressure Transient Loading; Revision 00

EC 346133; Replace RCIC Orifice 1E51-D307; Revision 0

EC 346528; Provide Guidance for the Use of Intermittent Loads; Revision 0

EC 353221; SSPI Impact During AC Load Shed of Division 1 and 2 Battery Charger Tests; Revision 0

EC 356294; Energizing the Main and Backup 125 Vdc Battery Chargers for Division 1 or 2 Simultaneously for Testing; dated July 19, 2005

EC 356324; Evaluation of the Capability of RCIC Governor EGM/RGSC to Operate in Station Blackout Environment; dated July 17, 2005

EC 356326; Compute RCIC Room Temperature for Station Blackout Conditions; dated July 18, 2005

EC 356331; Owner’s Review of GE Dose Evaluation for Station Blackout; Revision 0

Issue Reports Generated Due to the Inspection

00348798; Damaged Insulation Located Above 250V Battery Room in Unit 1 Division 1; dated June 29, 2005

00349252; Editorial Error in Calculation VX-09: Assumption 8 Incorrect; dated June 30, 2005

00349271; Calculation D34, Revision 02A Typographical Error; dated June 29, 2005

00349538; NRC Noted Discrepancies in LGA-RI-101; dated July 1, 2005

00350613; Impact of Station Blackout Temperature on 250 Vdc Motor Control Center; dated July 6, 2005

00351039; Calculation D27 and D30 Load Duration Differences; dated July 7, 2005

00351884; Setpoint Calculation NED-I-EIC-0204 Minor Discrepancy; dated July 11, 2005

00352087; NRC 2005 Identified Calc D34, Rt1 and Rt2 Shown for One Way; dated July 12, 2005

00352118; Updated Final Safety Analysis Report, Table 3.2-1, Note 15, Requires Clarification; dated July 12, 2005

00352743; RCIC Pump NPSH Required at Rated Flow and Speed; dated July 13, 2005

00352944; LPCS Cubicle Calculation L-1024 Requires Revision; dated July 14, 2005

00353160; Calculation NED-I-EIC-0212 Minor Discrepancies; dated July 14, 2005

00353163; RCIC Electronic Governor Module Does Not Meet Station Blackout Qualification; dated July 14, 2005

00353539; Inadequate Documentation of Basis for Judgement; dated July 15, 2005

00353550; Error in GE Dose Evaluation for Change to RCIC Room; dated July 15, 2005

00354047; Error Found in Calculation ATD-0351, Revision 1; dated July 18, 2005

00354050; Evaluation of RCIC Water Leg Room Station Blackout Heatup Questioned; dated July 18, 2005

00354058; Questions Regarding Monitoring of RCIC Room Temperature; dated July 18, 2005

Issue Reports Reviewed During the Inspection

L1998-00356; Calculation Concerns Raised by Oversite Review in Closeout of Nuclear Tracking System Item; dated December 5, 1997

00162386; NOS Identified DC Ground Alarm Setpoint Inconsistent with LOR Procedure; dated November 8, 2001

00332170; DC Calculation Require Update (Historical); dated May 5, 2005

00341720; Abnormal Noise at Vacuum Pump; dated June 7, 2005

00342721; Check Valve Sticking Closed; dated June 10, 2005

00343067; U-1 RCIC Vacuum Pump Has Minor Shaft Leak; dated July 10, 2005

00346214; 241Y Feed to 235X and 235Y Trip; dated June 22, 2005

Letters

LaSalle County Station Unit 1 Initial Test Program - Special Test; dated April 16, 1981

Emergency Procedures and Training for Station Blackout Events; dated June 22, 1981

Proposed Amendments to Technical Specification for Facility Operating License NPF-11 and NPF-18 Diesel Generator Lube Oil Modification; dated August 23, 1985

Proposed Amendments to Technical Specification for Facility Operating License NPF-18 Diesel Generator Lube Oil Modification Revised Submittal; dated August 28, 1985

Proposed Amendments to Technical Specification for Facility Operating License NPF-11 Diesel Generator Lube Oil Modification Submittal; dated October 14, 1986

Proposed Amendments to Technical Specification for Facility Operating Licenses NPF-11 and NPF-18 - To Allow One Unit Operation with “0" DG Out of Service for Specified Required Surveillances; dated January 19, 1987

Response to Station Blackout Rule; dated April 17, 1989

Supplemental Response to Station Blackout Rule; dated March 30, 1990

Supplemental Response to NRC Bulletin 88-04 “Potential Safety-Related Pump Loss”; dated April 29, 1990

Revised Response to Station Blackout Rule; dated June 22, 1990

Unit 1 Division 2, 125V Battery Replacement; dated October 17, 1990

Unit 2 Division 2, 125V Battery Replacement; dated August 30, 1991

Supplemental Response to Station Blackout Rule; dated September 23, 1991

Safety Evaluation of the LaSalle County Station Response to the Station Blackout Rule; dated March 6, 1992

Response to Safety Evaluation on the Station Blackout Rule; dated May 15, 1992

Safety Evaluation Related to Station Blackout Analysis, LaSalle County Station, Units 1and 2; dated July 17, 1992

Station Blackout Rule (10 CFR 50.63) Implementation Status; dated September 2, 1993

Application for Amendment to Facility Operating Licenses, Appendix A, Technical Specification Section 3/4.8, Electrical Power Systems; dated June 8, 1995

Clarification to the Safety Evaluation of the LaSalle County Station Response to the Station Blackout Rule; dated May 28, 1997

Supplemental Safety Evaluation of the LaSalle County Station Response to the Station Blackout Rule; dated December 4, 1997

Response to Request for Additional Information License Amendment Request for Power Uprate Operation; dated February 23, 2000

Response to Request for Additional Information License Amendment Request for Power Uprate Operation; dated March 31, 2000

Licensee Event Reports

2-2003-007-00; TS 3.8.4 Violation Due To Common Mode Battery Charger Failures; dated August 6, 2003

315/1999-027-00; Underrated Fuses Used in 250 VDC System Could Result in Lack of Protective Coordination; dated November 29, 1999

Miscellaneous Documents

CHRON#302940; RCIC Turbine Exhaust High Pressure Trip Setpoint Change Setpoint Change Nos. S01-1-94-038 and S01-2--94-037; dated September 30, 1994

EBO-93-324; RCIC Turbine Exhaust Pressure Trip Setpoint Modification; dated September 3, 1993

EIC-94-049; LaSalle Station Units 1 and 2 - Calibration Setpoint for RCIC Turbine Exhaust High Pressure Trip; Revision 0

LS-NSLD-0039-1; Suppression Pool Temperature and Mass for RCIC Operation Following Station Blackout; dated May 24, 1990

LS-PMED-00095; Station Blackout; dated May 29, 1990

LS-PMED-0151-01; Input for NPSH Limit Calculation EWR 92-123; dated July 20, 1992

NEDC-32701P; Power Uprate Safety Analysis Report for LaSalle County Station, Units 1 and 2; Revision 2

NEDE-22017; BWR Owners' Group Evaluation of RCIC Turbine Exhaust Pressure Trip for LOCA Applications; dated November 1981

SEAG 03-000122; Unit 2 DC Ground Recorder/Alarm Calibrations (Work Order 99207083); dated June 4, 2002

SEAG 03-000109; DC System Grounds Task Force Report; dated May 25, 2005

SEAG 05-000069; LaSalle Special Test of RCIC Electronic Controls EG-M/RGSC; dated July 16, 2005

SIL No. 475R2; GE Nuclear Services Information Letters: RCIC and HPCI High Steam Flow Analytic Limit; dated November 28, 1988

Effectiveness Review AT 175593-37; dated May 5, 2005

Focused Area Self-Assessment Report, Readiness Assessment for NRC SSDPC Inspection of the DC and RCIC Systems, and Components in Other Systems; dated May 27, 2005

LaSalle County FAC Program Basis Document; Revision 7

List of Deferred RCIC Preventive Maintenance; dated June 30, 2005

List of EQ Related Equipment in RCIC Room; dated June 30, 2005

List of RCIC Preventive Maintenance Changes; dated July 11, 2005

Maintenance History for Battery Chargers; dated June 30, 2005

Operations Training Program - DC Distribution; dated May 2, 2005

Operations Training Program - Reactor Core Isolation Cooling System; dated November 10, 2003

RCIC Work History Report; dated June 30, 2005

Root Cause Investigation of Unit 1 RCIC Suction Pressure Transient During Full Flow Test; dated March 25, 2004

Specification Sheet for Oil Cooler for Type GS Turbine; dated March 12, 1970

Specification Sheet for Reactor Core Isolation Cooling System; dated August 2, 1971

<u>Operability Determinations</u>

01-007; Unit 2 RCIC Barometric Condenser Condensate Pump; dated February 21, 2001

03-015; Unit 1 RCIC Pump (1E51-C001) Suction Piping; dated March 25, 2004

05-005; Reactor Core Isolation Cooling NPSH; dated July 14 2005

05-006; Units 1 and 2 Operable - Governor EGM Not Qualified; dated July 14, 2005

<u>Operating Experience</u>

IN 1988-72; Inadequacies in the Design of DC Motor Operated Valves; dated September 2, 1988

IN 1991-51; Inadequate Fuse Control Programs; dated August 20, 1991

Procedures

CC-AA-308; Control and Tracking of Electrical Load Changes; Revision 4

CY-AB-120-200; Storage Tanks Chemistry; Revision 4

ER-AA-430; Conduct of Flow Accelerated Corrosion Activities; Revision 1

ER-AA-430-1001; Guidelines for Flow Accelerated Corrosion Activities; Revision 1

LES-DC-103A; Division 1 Battery Charger Capacity Test; Revision 12

LGA-RI-101; Unit 1 Alternate Vessel Injection Using RCIC Including Defeat of RCIC Isolations; Revision 1

LOA-FLD-001; Flooding; Revision 6

LOA-AP-101; Unit 1, AC Power System Abnormal; Revision 20

LOA-AP-201; Attachment K: Station Black-out Contingencies; Revision 15

LOA-AP-201; Attachment N: DC Load Shedding; Revision 15

LOA-DC-201; Unit 2 DC Power System Failure; Revision 7

LOP-DC-05; 125 Vdc System Division 2 Ground Location and Isolation; Revision 21

LOP-DC-07; Battery Equalizing Charges; Revision 30

LOP-R1-01; Filling, Venting, and Draining of Reactor Core Isolation Cooling System; Revision 27

LOR-1H13-P601-D203; RCIC Turbine Bearing High Temperature; Revision 2

LOR-2PM01J-B504; 125 Vdc Panel 212X/Y Ground Detector Alarm; Revision 1

LOS-AA-S101; Unit 1 Shiftly Surveillance for Mode 1, 2, or 3 Attachment A; Revision 31

OU-AA-103; Shutdown Safety Management Program; Revision 4

WC-AA-106; Attachment 4, Job Types; Revision 2

Surveillances

L1R10-RT-032-37; 1-R1-001: Supply Steam Line Drain RT Data Sheet; dated January 11, 2005

L2R10-RT-053-56; 2-R1-001: Supply Steam Line Drain RT Data Sheet; dated January 28, 2005

LES-DC-101A; Division 1 125 Volt Battery Inspection For Units 1 and 2; dated April 2, 2004

LES-DC-101B; Division 2 125 Volt Battery Inspection For Units 1 and 2; dated January 4, 2005

LES-DC-101C; Division 3 125 Volt Battery Inspection; dated May 16, 2005

LIS-RI-103A; Unit 1 RCIC Equipment Room/Steam Line Tunnel High Ambient and Differential Temperature Outboard Isolation (Div 1) Calibration; dated January 31, 2004

LIS-RI-113; Unit 1 RCIC Pump Water Leg Line Low Pressure Calibration; dated December 19, 2003

LIS-RI-202; Unit 2 RCIC Pump Discharge Flow Indication Calibration; dated March 22, 2004

LIS-RI-209; RCIC Turbine Exhaust Diaphragm High Pressure Isolation; dated April 16, 2005

LIS-RI-214; Unit 2 RCIC Pump Suction Line High Pressure Calibration; dated February 26, 2004

LIS-RI-215; Unit 2 RCIC Control System Calibration; dated January 19, 2005

LIS-RI-313; Unit 1 RCIC Pump Water Leg Line Low Pressure Functional Test; dated June 7, 2005

LIS-RI-316; Unit 1 Cycled Condensate Storage Tank Low Level RCIC Suction Fun; dated May 5, 2005

LIS-RI-412; Unit 2 Reactor Vessel High Water Level 8 RCIC Turbine Trip and Main Turbine Trip/Feedwater Pump Trip Functional Test; dated April 19, 2005 and January 4, 2005

LIS-RI-413; Unit 2 RCIC Pump Water Leg Line Low Pressure Functional Test; dated June 20, 2005

LIS-RI-416; Unit 2 Cycled Condensate Storage Tank Low Level RCIC Suction Functional Test; dated June 20, 2005

LOS-DC-Q2; Division 1 125 Vdc Battery; dated February 31, 2005

LOS-RI-Q1; Unit 2 Reactor Core Isolation Cooling System Valve Inservice Test; dated May 22, 2005

LOS-RI-Q5; Reactor Core Isolation Cooling (RCIC) System Pump Operability, Valve Inservice Tests in Modes 1, 2, 3, and Cold Quick Start; dated June 10, 2005

LOS-RI-R3; RCIC Operability; dated February 11, 2004

LTS-700-6; Unit 2 125V Battery Division 1 Service Discharge Test; dated January 23, 2004 and February 21, 2005

LTS-700-7; Unit 1(2) Division II Battery Service Test Discharge; dated February 6, 2004

LTS-700-8; Unit 1(2) Division III Battery Service Test Discharge; dated January 16, 2004

LTS-900-5; Reactor Core Isolation Cooling Head Spray Pressure Isolation Check Valves Water Leak Test 1(2)E51-F065; dated February 22, 2005

LTS-700-19; Unit 1(2) Division II 125V Battery Modified Performance Test; dated February 27, 2005

LTS-700-20; Unit 1(2) Division III Battery Modified Performance Test; dated February 4, 2005

Work Orders

484259-01; Unit 2 DC Motor Operated Globe Valve Calculation LAS-1E51-F046; dated January 18, 2005

557075; Unit 1 DC Motor Operated Globe Valve Calculation LAS-1E51-F046; dated December 7, 2004

594676; ES 1VY04A LPCS Pump Rm LTS-200-19 Air Side Flowrate Test; dated April 4, 2005

615044; ES 1VY04A Water Flowrate Test LTS-200-12; dated October 18, 2004

633872; LOS-ZZ-A2 Winterize Station ATT A; dated May 16, 2005

722105; LOS-ZZ-A2 Preparation for Summer Operations AAT B; dated April 25, 2005

Vendor Manuals

J-0246-000; VETIP Manual for Westinghouse Type HFB Breaker

SCI # B5-CC2000-XX; Vendor Manual for SCI Battery Chargers; dated March 9, 2000

Pressure and Vacuum Switches for Process; dated April 1993

Solid-State Digital Timer Model SST-2; 1990

LIST OF ACRONYMS USED

AC	Alternating Current
ADAMS	Agencywide Documents Access and Management System
ADS	Automatic Depressurization System
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
DC	Direct Current
DRS	Division of Reactor Safety
ECR	Engineering Change Request
EGM	Electronic Governor Module
IMC	Inspection Manual Chapter
IEEE	Institute of Electrical and Electronics Engineers
IR	Issue Report, as used by licensee (and NRC Inspection Report)
LOCA	Loss of Coolant Accident
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
OE	Operability Evaluation
RCIC	Reactor Core Isolation Cooling
RGSC	Ramp Generator Signal Converter
RPM	Revolutions Per Minute
SBO	Station Blackout
SDP	Significance Determination Process
SSDPC	Safety System Design and Performance Capability
UFSAR	Updated Final Safety Analysis Report