March 30, 2007

Mr. Fred R. Dacimo
Site Vice President
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT 2 - NRC COMPONENT DESIGN BASES INSPECTION REPORT 05000247/2007007

Dear Mr. Dacimo:

On February 15, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Indian Point Nuclear Generating Unit 2. The enclosed inspection report documents the inspection results, which were discussed on February 15, 2007, with Messrs. J. Ventosa and J. Comiotes and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. This particular inspection was performed by a team of NRC inspectors and contractors using NRC Inspection Procedure 71111.21, "Component Design Bases Inspection." In conducting the inspection, the team examined the adequacy of selected components and operator actions to mitigate postulated transients, initiating events, and design basis accidents. The inspection also reviewed Entergy's response to selected operating experience issues. The inspection involved field walkdowns, examination of selected procedures, calculations and records, and interviews with station personnel.

This report documents eight NRC-identified findings that were of very low safety significance (Green). Seven of these findings were determined to involve violations of NRC requirements. However, because of the very low safety significance of the violations and because they were entered into your corrective action program, the NRC is treating the violations as non-cited violations (NCV) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV in this report, 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, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspectors at Indian Point Unit 2.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records 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/

Lawrence T. Doerflein, Chief
Engineering Branch 2
Division of Reactor Safety

Docket No. 50-247
License No. DPR-26

Enclosure: Inspection Report 05000247/2007007

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-247
License No.	DPR-26
Report No.	05000247/2007007
Licensee:	Entergy Nuclear Northeast
Facility:	Indian Point Nuclear Generating Unit 2
Location:	450 Broadway, GSB Buchanan, NY 10511-0308
Dates:	January 8 to February 15, 2007
Inspectors:	S. Pindale, Senior Reactor Inspector (Team Leader) D. Orr, Senior Reactor Inspector J. Krafty, Reactor Inspector J. Lilliendahl, Reactor Inspector H. Anderson, NRC Mechanical Contractor S. Kobylarz, NRC Electrical Contractor W. Sherbin, NRC Mechanical Contractor S. Smith, Nuclear Safety Professional Development Program (Observer)
Approved by:	Lawrence T. Doerflein, Chief Engineering Branch 2 Division of Reactor Safety

Enclosure

TABLE OF CONTENTS

EXECUTIVE SUMMARY

During the period from January 8 through February 15, 2007, the U.S. Nuclear Regulatory Commission (NRC) conducted a team inspection at the Indian Point Nuclear Generating Unit 2 (IP 2) in accordance Inspection Procedure 71111.21, "Component Design Bases Inspection." The inspection procedure is conducted biennially as part of the NRC's Reactor Oversight Process (ROP).[1] The objective of the inspection was to verify that the IP 2 design bases had been correctly implemented for selected risk-significant components, and that operating procedures and operator actions were consistent with the design and licensing bases. This was to ensure that the selected components were capable of performing their intended safety functions and could support the proper operation of the associated systems. The inspection team consisted of seven inspectors, including a team leader and three inspectors from the NRC's Region I Office, and three contractors. The inspection involved four weeks of on-site effort.

The team selected nineteen components for a detailed design review after completing a detailed selection process. In selecting samples for review, the team focused on those components and operator actions that have a high relative contribution to the risk of a postulated core damage accident if the component was to fail or if the operator did not successfully complete the action. The team also assessed available margin for the risk-significant components in selecting the samples. The selected samples included components in the safety injection (SI), residual heat removal (RHR), auxiliary feedwater (AFW), onsite electrical power, and off-site electrical power systems. The team selected five risk-significant operator actions for review using the complexity of the action, time to complete the action, and extent of training on the action as inputs. The team also selected six operating experience issues related to the selected components or generic issues to verify they had been appropriately assessed and dispositioned. For each sample selected, the team reviewed design calculations, corrective action reports, maintenance and modification histories, and associated operating and testing procedures. The team also performed walkdowns of the accessible components to assess their material condition.

Overall, the inspection team determined that the components reviewed were capable of performing their intended safety functions. The team also found that the operating procedures, operator training and equipment staging adequately supported completion of the operator actions and were consistent with the design and licensing bases. The team did identify eight findings of very low safety significance (Green) and one unresolved item. The eight findings are listed in the "Summary of Findings" section of this report. The team assessed the safety significance of each of the findings using the NRC's Significance Determination Process (SDP).[2] Also, for each of the findings where current operability was a relevant question, Entergy completed an operability evaluation. In each case, Entergy determined the equipment was operable. The inspection team independently confirmed Entergy's conclusions. All of the findings were entered into Entergy's corrective action program to ensure a more comprehensive assessment of the issue and to identify and implement appropriate corrective actions.

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[1] Described in NRC's Inspection Manual Chapter 0308, Reactor Oversight Process

[2] Described in NRC's Inspection Manual Chapter 0609, Appendix A, Determining the Significance of Reactor Inspection Findings for At-Power Situations

Under the NRC's Reactor Oversight Process, findings of very low safety significance (Green) are addressed through the facility's corrective action program. Future NRC inspections, most notably the biennial Problem Identification and Resolution (PI&R) team Inspection, review a substantial sample of Entergy's response to the Green findings and assess the adequacy of the actions taken to correct the deficiencies.

The findings are also an input into the NRC's assessment process.[3] The most recent assessment of IP 2 issued on March 2, 2007 (ADAMS Ref. ML070610603), concluded that the plant's performance was in the Licensee Response Column of the NRC's Action Matrix. Because the findings of this Component Design Bases Inspection were all Green, the NRC's overall assessment of IP 2 will not change from the Licensee Response Column as a result of this inspection. The recent assessment also identified a substantive cross-cutting issue in the area of human performance regarding procedure adequacy. The Reactor Oversight Process considers that the areas of human performance, problem identification and resolution and safety conscious work environment, contain performance attributes that extend across (cross-cut) all areas of reactor plant operation. As noted in the inspection report, several of the findings had cross-cutting aspects. As part of the assessment process, the NRC performs a collective review semi-annually of cross-cutting aspects of all inspection results from the previous twelve months, and monitors and evaluates a plant licensee's actions to address a substantive cross-cutting issue.

This inspection is a key part of NRC's inspection effort to assure overall plant safety, protection of the public and the environment, and efficacy of key plant design features and procedures. Many other NRC inspection and review activities are also important to NRC's role of ensuring safety. More detail is provided in the NRC's description of the Reactor Oversight Process at http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/index.html. A similar inspection is planned for the Indian Point Nuclear Generating Unit 3 in the Fall of 2007.

[3] As described in Inspection Manual Chapter 0305, Operating Reactor Assessment Program

SUMMARY OF FINDINGS

IR 05000247/2007007; 1/8/2007 - 2/15/2007; Indian Point Nuclear Generating Unit 2; Component Design Bases Inspection.

This inspection covers the Component Design Bases Inspection, conducted by a team of four NRC inspectors and three NRC contractors. Eight findings of very low safety significance (Green) were identified, seven of which involved a violation of regulatory requirements and are considered to be non-cited violations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The team identified a finding of very low significance involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," in that, Entergy did not ensure adequate suction submergence for the three safety injection (SI) pumps by not properly translating vortex and net positive suction head (NPSH) design parameters into calculations relative to reactor water storage tank (RWST) level. Specifically, Entergy used a non-conservative method to calculate the level required to prevent pump vortexing, and used a non-conservative RWST level value for determining available NPSH for the SI pumps. Entergy entered the issue into their corrective action program and revised the affected calculations.

  The finding is more than minor because the calculation deficiencies represented reasonable doubt on the operability of the SI pumps, even though the pumps were ultimately shown to be operable. The finding is associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding has very low safety significance, based on Phase 1 of the significance determination process (SDP), documented in NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," because it was a design deficiency that did not result in a loss of SI system operability, based upon the team's verification of Entergy's revised calculations. (Section 1R21.2.1.1)

- Green. The team identified a finding of very low significance involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," in that, Entergy did not accurately incorporate design parameters into valve thrust calculations for motor operated valve (MOV) 746 and MOV 747. Specifically, Entergy used an incorrect and non-conservative differential pressure in the calculations for MOV 746 and MOV 747, which were developed to verify that the valves could develop sufficient thrust to open under postulated design basis conditions. Additionally, an incorrect equation was used in determining the reduction in motor torque due to degraded voltage conditions.

Entergy entered the issue into their corrective action program and revised the affected calculations using the correct information.

The finding is more than minor because the calculation deficiencies represented reasonable doubt on the operability of MOV 746 and MOV 747. The finding is associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding has very low safety significance, based on Phase 1 of the SDP, because it was a design deficiency that did not to result in a loss of MOV 746 and MOV 747 operability, based upon the team's verification of Entergy's revised calculations. (Section 1R21.2.1.2.b1)

- Green. The team identified a finding of very low safety significance involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," in that, Entergy did not establish adequate design control measures to ensure the availability of the turbine driven auxiliary feedwater pump (TDAFWP) during a postulated loss-of-offsite power (LOOP) event. Under certain LOOP situations, the team determined that the TDAFWP steam supply could be inadvertently isolated because of inadequate calculations and procedures for limiting the AFWP room temperature rise. Specifically, a calculation to determine the auxiliary feedwater pump (AFWP) room temperature rise during a LOOP did not include heat input from the TDAFWP. Further, actions that could limit the rise in AFWP room temperature and prevent the inadvertent isolation of the TDAFW pump (opening an AFWP room roll-up door or promptly restoring forced ventilation) were not included in procedures. Entergy entered this issue into their corrective action program, implemented immediate compensatory actions, and revised AFWP room temperature rise calculations.

  The finding is more than minor because it is associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The finding has very low safety significance, based on Phase 1 of the SDP, because it did not represent the loss of safety function of the TDAFWP (single train) for greater than its 72 hour technical specification allowed outage time, based on the team's review and assessment of site ambient temperature data over the last year. (Section 1R21.2.1.7.b)

- Green. The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50.65(a)(1), the Maintenance Rule, in that, Entergy failed to monitor the gas turbine (GT) system in a manner that provided reasonable assurance that the system could perform its intended safety function. Specifically, Entergy did not establish appropriate GT reliability goals, and therefore did not take corrective actions, when GT-1 had exceeded these goals for maintenance preventable functions failures (MPFF). In addition, Entergy did not properly classify repeat MPFFs, which resulted in a similar failure to take corrective actions as required. This resulted in additional GT-1 out of service time that would not have happened if appropriate actions had been taken. Entergy entered this issue into their corrective action program and lowered the allowable goal for MPFFs, and revised the GT-1 (a)(1) action plan to improve reliability.

The finding is more than minor because appropriate GT reliability goals were not established commensurate with safety and appropriate corrective actions were not taken when goals were not met. This finding is associated with the equipment performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The finding has very low safety significance, based on Phase 1 and Phase 2 of the SDP, which considered that the additional GT-1 out of service time due to this issue could be as much as three days. The finding has a cross-cutting aspect in the area of human performance because Entergy did not adequately ensure procedures were complete, accurate, and up-to-date. Specifically, procedure ENN-DC-171, "Maintenance Rule Monitoring," did not provide steps to discriminate between the classification of an initial design deficiency and further failures due to the same condition, resulting in mis-classifying several GT functional failures. (1R21.2.1.10.b1)

- <u>Green</u>. The team identified a finding of very low safety significance involving Entergy procedure, EN-LI-102, "Corrective Action Process," in that, Entergy failed to take corrective actions to address degraded GT-1 reliability. This resulted in a two and one half day time period in January 2007 when GT-1 and GT-3 were simultaneously inoperable because, after GT-3 was made inoperable for planned maintenance activities, GT-1 was subsequently found to be inoperable. Specifically, the reliability of GT-1 declined from an average of 75% for 2005 and the first 10 months of 2006, to 50% for the three months from November 2006 to January 2007; however, Entergy did not take actions to correct this degraded reliability. Entergy entered this issue into their corrective action program and developed an action plan to address GT reliability issues.

  The issue is more than minor because it is associated with the equipment reliability attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding has very low safety significance, based on Phase 1 and Phase 2 of the SDP, assuming that both GT-1 and GT-3 were unavailable for the two and one half days, due to this issue. The finding has a cross-cutting aspect in the area of problem identification and resolution because Entergy did not correct degraded reliability of GT-1, resulting in having GT-1 and GT-3 simultaneously inoperable. (1R21.2.1.10.b2)

- <u>Green</u>. The team identified a finding of very low safety significance (Green) involving a non-cited violation of Technical Specification 3.8.6.6, in that, Entergy did not perform station battery capacity testing in accordance with IEEE Standard 450-1995 (related to battery maintenance and testing). Specifically, Entergy procedurally terminated battery capacity testing at the rated discharge time (four hours), before reaching the minimum voltage, as specified by IEEE Standard 450-1995. This prevented accurate quantitative measurement of capacity degradation and identification of the need to conduct potential accelerated battery testing, as specified by both IEEE Standard 450-1995 and the technical specifications, if battery capacity drops by more than 10% relative to the previous test. Entergy entered the issue into their corrective action program and performed calculations using past test data, which demonstrated that the capacities of station batteries had not degraded more than 10%.

This issue is more than minor because it is associated with the procedure quality attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding has very low safety significance, based on Phase 1 of the SDP, because it did not represent the loss of station battery safety function, based upon the team's verification of Entergy's calculations. (Section 1R21.2.1.13.b1)

- Green. The team identified a finding of very low safety significance involving a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," in that, Entergy did not take effective corrective actions for a condition adverse to quality concerning out-of-tolerance inter-tier resistances on the No. 21 station battery. Specifically, after repeated failures of the No. 21 station battery inter-tier resistance testing, vendor and IEEE Standard 450-1995 recommended corrective actions were not taken to correct the adverse out-of-tolerance resistance trend. Entergy entered the issue into their corrective action program and performed calculations, which demonstrated that the voltage drop due to the as-found resistance of the inter-tier connections was small and did not impact No. 21 battery operability.

  This issue is more than minor because if it was left uncorrected, it would have become a more significant safety concern. Specifically, high resistance connections in a battery that is loaded during accident conditions can cause localized heating and can cause permanent damage to the battery. The finding has very low safety significance, based on Phase 1 of the SDP, because it did not represent the loss of No. 21 station battery safety function, based upon the team's verification of Entergy's revised calculations. The finding has a cross-cutting aspect in the area of problem identification and resolution because Entergy did not take effective corrective actions to address the adverse trend of out-of-tolerance inter-tier resistances. (Section 1R21.2.1.13.b2)

- Green. The team identified a finding of very low safety significance involving a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," in that, Entergy did not promptly identify and correct a condition adverse to quality, with respect to known errors in the No. 23 station battery design calculations. Specifically, Entergy did not recognize at the appropriate time the need to write a condition report, perform an operability determination, or place controls on the use of the No. 23 battery design calculations when errors were discovered in the No. 23 battery design calculations that significantly lowered the battery capacity margin. Entergy entered the issue into their corrective action program and performed calculations, which demonstrated No. 23 station battery operability through the next refueling outage, based on the calculated margin and conservatisms available.

  This issue is more than minor because it is associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding has very low safety significance, based on Phase 1 of the SDP, because it did not represent the loss of No. 23 station battery safety function, based upon the team's verification of Entergy's revised calculations.

The finding has a cross-cutting aspect in the area of problem identification and resolution because Entergy failed to promptly identify the decrease in margin found in the No. 23 battery design calculations of record. (Section 1R21.2.1.13.b3)

B. Licensee-identified Violations

None.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R21 Component Design Bases Inspection (IP 71111.21)

.1 Inspection Sample Selection Process

The team selected risk significant components and operator actions for review using information contained in the Indian Point 2 Probabilistic Risk Assessment (PRA) and the Nuclear Regulatory Commission's (NRC) Standardized Plant Analysis Risk (SPAR) model. Additionally, the Indian Point 2 Significance Determination Process (SDP) Phase 2 Notebook, Revision 2, was referenced in the selection of potential components and actions for review. In general, the selection process focused on components and operator actions that had a risk achievement worth (RAW)[1] factor greater than 2.0 or a Risk Reduction Worth (RRW)[2] factor greater than 1.005. The components selected were located within both safety related and non-safety related systems, and included a variety of components such as pumps, valves, diesel generators, transformers, and electrical buses.

The team initially compiled an extensive list of components based on the risk factors previously mentioned. The team performed a margin assessment to narrow the focus of the inspection to 19 components and five operator actions. The team's evaluation of possible low design margin considered original design issues, margin reductions due to modifications, or margin reductions identified as a result of material condition/equipment reliability issues. The margin assessment evaluated the impact of licensing basis changes that could reduce safety analysis margins. The assessment also included items such as failed performance test results, corrective action history, repeated maintenance, maintenance rule (a)(1) status, operability reviews for degraded conditions, NRC resident inspector input of equipment problems, plant personnel input of equipment issues, system health reports and industry operating experience. Consideration was also given to the uniqueness and complexity of the design and the available defense-in-depth margins. The margin review of operator actions included complexity of the action, time to complete action and extent of training on the action.

This inspection effort included walk-downs of selected components, a review of selected simulator scenarios, interviews with operators, system engineers and design engineers, and reviews of associated design documents and calculations to assess the adequacy of the components to meet both design bases and risk informed beyond design basis requirements. A summary of the reviews performed for each component, operator

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[1]RAW is the factor by which the plant's core damage frequency increases if the component or operator action is assumed to fail.

[2]RRW is the factor by which the plant's core damage frequency decreases if the component or operator action is assumed to be successful.

action, operating experience sample, and the specific inspection findings identified are discussed in the following sections of the report. Documents reviewed for this inspection are listed in the attachment.

.2 Results of Detailed Reviews

.2.1 Detailed Component Design Reviews (19 Samples)

.2.1.1 No. 21 Safety Injection Pump

a. Inspection Scope

The team reviewed design basis documents, including hydraulic calculations, technical specifications, accident analyses and drawings to ensure the No. 21 safety injection (SI) pump was capable of meeting system functional and design basis requirements. Because the water source for the pump during the injection phase of a postulated accident is the refueling water storage tank (RWST), the tank level setpoints and uncertainty calculations were reviewed. The team also reviewed SI pump test results, system health reports, and corrective action documents to verify SI pump design margins were being maintained and to confirm that Entergy was entering problems that could affect system performance into the corrective action program. The team reviewed operating and emergency procedures to verify adequate RWST inventory existed to inject water into the reactor during a postulated accident, and to ensure pump suction swapover occurred before the onset of vortexing at the RWST inlet piping. To assess the general condition of the pump, the team performed walkdowns of the SI pump area. The team also reviewed SI pump and motor cooler systems and SI pump minimum flow requirements to assess the ability of the SI pump to operate under design basis conditions.

b. Findings

Introduction: The team identified a finding of very low significance (Green) involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, “Design Control,” in that, Entergy did not ensure adequate suction submergence for the three safety injection (SI) pumps by not properly translating vortex and net positive suction head (NPSH) design parameters into calculations relative to RWST level. Specifically, Entergy used a non-conservative method to calculate the reactor water storage tank (RWST) level required to prevent pump vortexing and used a non-conservative level value for determining available NPSH for the SI pumps.

Description: There are numerous methodologies available to calculate the minimum submergence level to prevent vortexing associated with pumps, primarily based on correlations of experimental data. The team noted that the methodology used in calculation FMX-00085, “Minimum Submergence Level SI/RHR and Containment Spray,” Revision 0, to determine the minimum height of water above the SI pumps’ intake to preclude vortex formation in the RWST was not appropriate. Specifically, the onset of vortexing was calculated using a methodology which was based upon fluid withdrawal from a tank at a constant level. The team questioned the validity and application of this approach, because the RWST is not maintained at a constant level during the postulated scenario, but rather, it would be pumped down and level would decrease.

Based upon the above concern, Entergy acknowledged that the methodology used was not appropriate for determining the onset of vortexing in the RWST, and entered the issue into the corrective action program (CR-IP2-2007-00409 & CR-IP2-00439) to evaluate other calculational methods to ensure that the onset of vortexing would not occur prior to suction swapover for the SI pumps (from the RWST to the containment sump). The results of these other methods confirmed that the original methodology was non-conservative, resulting in reducing the available margin in RWST level from about 5 inches to as low as 2.5 inches. The team determined that this design control deficiency did not result in a loss of safety function of the SI pumps because there was still adequate submergence to prevent vortexing at the suction inlet piping in the RWST.

The team also reviewed Calculation FMX-00050-01, "SI Pump Available NPSH," Revision 1. The team noted that the water level in the RWST corresponding to the beginning of the pump suction swapover sequence was used in the calculation as the available static head of water (about 90 feet plant elevation). Since the SI pumps could be operating at a RWST water level corresponding to the level where operators terminate pump operation (about 82.5 feet), the calculation should have used 82.5 feet elevation when determining the static head of water. This resulted in reducing the available pump NPSH by about 7 feet. Entergy determined, and the team confirmed, that adequate NPSH remained for the SI pumps, but the margin over the required NPSH was reduced. Entergy entered this issue in the corrective action program (CR-IP2-00712).

Entergy's corrective actions for this issue included recalculating the SI pump vortex limit using the appropriate methodologies, and they determined that there was not an operability issue. With respect to SI pump NPSH, Entergy confirmed that the available NPSH remained sufficient to prevent degraded pump performance and did not adversely affect pump operability. Further, Entergy plans to address the extent of condition as part of the condition report evaluations. The team reviewed Entergy's corrective actions and found them to be appropriate.

<u>Analysis</u>: The team determined that Entergy's failure to ensure adequate suction submergence for the three SI pumps by not properly translating vortex and NPSH design parameters into calculations relative to RWST level was a performance deficiency that was reasonably within Entergy's ability to foresee and prevent. Specifically, Entergy used a non-conservative method to calculate the level required to prevent pump vortexing and used a non-conservative RWST level value for determining available NPSH for the SI pumps.

The finding was more than minor because it was similar to NRC Inspection Manual Chapter 0612, Appendix E, "Examples of Minor Issues," Example 3.j, in that, calculation deficiencies represented reasonable doubt on the operability of the SI pumps. The finding was associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In particular, the formation of vortexing at the intake of the SI suction line could result in air entrainment, which could cause pulsating pump flow and/or degradation in pump performance; and inadequate pump NPSH available could result in pump cavitation and reduced flow. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the team conducted a Phase 1 SDP screening and determined the finding was of very low safety significance (Green) because it was a design deficiency that did not result in a loss of SI system operability, based upon the team's verification of Entergy's evaluation.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” requires, in part, that measures shall be established to ensure that the design basis for structures, systems, and components are correctly translated into specifications, drawings, procedures, and instructions. Contrary to this requirement, as of January 24, 2007, Entergy had not correctly translated design bases into specifications, drawings, procedures, and instructions when they used a non-conservative methodology for calculating the onset of vortexing at the intake of the SI pump common suction line from the RWST, resulting in reducing the available margin in RWST level from about 5 inches to as low as 2.5 inches. Additionally, Entergy did not translate the appropriate RWST water level when calculating the NPSH available to the SI pumps, resulting in reducing the available pump NPSH by about 7 feet. Because this violation is of very low significance and has been entered into Entergy's corrective action program (CR-IP2-2007-00409, CR-IP2-2007-00439, and CR-IP2-2007-00712), this violation is being treated as a non-cited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000247/2007007-01, Inadequate Design Control Associated with Vortexing and Net Positive Suction Head Calculations)**

.2.1.2 No. 21 Residual Heat Removal Heat Exchanger Discharge Valve (MOV 747)

a. Inspection Scope

The team selected residual heat removal (RHR) heat exchanger discharge motor operated valve (MOV) 747 as a representative high risk valve sample. The team reviewed calculations, procedures, periodic verification test results and technical reports to verify the valve's capability to perform during postulated design basis accident conditions. The team also interviewed engineers and reviewed correspondence related to NRC Generic Letter 96-05, “Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves,” to verify that Entergy was meeting its commitments for MOV periodic verification. Preventive maintenance requirements and corrective action reports were reviewed in order to determine the performance and operational history of the valve.

b. Findings

1. Inadequate Differential Pressure Value Used to Ensure MOV Capability

Introduction: The team identified a finding of very low significance (Green) involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," in that, Entergy did not accurately incorporate design parameters into valve thrust calculations for motor operated valve (MOV) 746 and MOV 747. Specifically, Entergy used an incorrect and non-conservative differential pressure in the calculations for MOV 746 and MOV 747, which were developed to verify that the valves could develop sufficient thrust to open under postulated design basis conditions. Additionally, an incorrect equation was used in determining the reduction in motor torque due to degraded voltage conditions. The equation used was only valid for degraded voltages 70% or greater than nominal voltage; however, the assumed degraded voltage for MOV 746 and MOV 747 was less than 70% of nominal voltage.

Description: The team noted that calculation PGI-00059-02, "746 & 747 Differential Pressure Calculation," revised the maximum differential pressure (to open) for MOV 747 from 1818 pounds per square inch - differential (psid) to 1600 psid (revised on October 24, 1997). The valve receives a signal to open upon the initiation of an accident signal, and the valve must overcome a differential pressure equivalent to the difference between reactor coolant system pressure and the refueling water storage tank head pressure (reverse to normal flow direction). The maximum differential pressure was based on a postulated large break loss-of-coolant accident (LOCA). However, the team questioned whether the worst case differential pressure would occur during a postulated small break LOCA, with its associated slower decrease in reactor pressure. Subsequently, Entergy confirmed that a more limiting differential pressure of 1852 psid should have been applied in the calculation and was consistent with the differential pressure assumed for a small break LOCA. This deficiency also applied to MOV 746 (the No. 22 RHR heat exchanger discharge valve). Direct substitution of the higher differential pressure into the calculation yielded a negative thrust margin. Therefore, Entergy performed a calculation for both MOV 746 and 747, using an as-tested dynamic stem coefficient, and showed that both valves still had sufficient thrust margin, and therefore remained capable of performing their intended design basis function.

The team also noted that the degraded voltage calculation for MOV 747 performed in MMS-00088, "Analysis of Thrust and Torque Limits for Motor Operated Valve 747," used an equation that was valid for degraded voltages greater than or equal to 70% of nominal voltage; however, the assumed degraded voltage for the valve was less than 70% of nominal voltage. Again, this deficiency similarly applied to MOV 746. Entergy performed a calculation using an appropriate methodology for determining actuator output torque, and determined that both valves still had sufficient margin and remained operable.

Entergy's corrective actions included performing calculations as discussed above and conducting associated operability assessments. The team reviewed the calculations for both issues as well as Entergy's associated operability assessments, and found them to be adequate. The team confirmed that the collective impact of the two deficiencies did not adversely impact the operability of the valves.

Analysis: The team determined that Entergy's failure to accurately incorporate design parameters into calculations for MOV 746 and MOV 747 was a performance deficiency that was reasonably within Entergy's ability to foresee and prevent. Specifically, Entergy used an incorrect and non-conservative differential pressure in the calculations for MOV 746 and MOV 747, which were developed to verify that the valves could develop sufficient thrust to open under postulated design basis conditions. Additionally, an incorrect equation was used in determining the reduction in motor torque due to degraded voltage conditions. The equation used was only valid for degraded voltages 70% or greater than nominal voltage; however, the assumed degraded voltage for MOV 746 and MOV 747 was less than 70% of nominal voltage.

The finding was more than minor because it was similar to NRC Inspection Manual Chapter 0612, Appendix E, "Examples of Minor Issues," Example 3.j, in that, calculation deficiencies represented reasonable doubt on the operability of MOV 746 and MOV 747. The finding was associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the team conducted a Phase 1 SDP screening and determined the finding was of very low safety significance (Green) because it was a design deficiency that was confirmed not to result in a loss of MOV 746 and MOV 747 operability, based upon the team's verification of Entergy's revised calculations.

Enforcement: 10 CFR 50 Appendix B, Criterion III, "Design Control," requires, in part, that measures shall be established to ensure that the design basis for structures, systems, and components are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, as of October 24, 1997, Entergy did not ensure that the design basis differential pressure for MOV 746 and MOV 747 was correctly translated into the appropriate valve calculations, when calculation PGI-00059 was revised to include an incorrect and non-conservative differential pressure. Additionally, Entergy did not ensure that an appropriate equation was used in determining the reduction in motor torque due to degraded voltage conditions for MOV 746 and MOV 747. Because this violation is of very low safety significance and has been entered into Entergy's corrective action program (CR-IP2-2007-00463), this violation is being treated as a non-cited violation consistent with Section VI.A.1. of the NRC Enforcement Policy. **(NCV 05000247/2007007-02, Inadequate Differential Pressure Value Used for MOV 746 and MOV 747 to Ensure Valve Capability)**

2. Use of Motor Control Center Methodology for MOV Periodic Verification

The team identified an unresolved item (URI) concerning the adequacy of the motor control center (MCC) testing methodology used for periodic verification of the design bases capability of safety-related MOVs. Entergy implemented MCC testing in 2004 as a method of implementing periodic verification in addition to the previously NRC-reviewed method of taking stem thrust and torque measurements at the valve. The MCC method uses motor current, voltage, and winding resistance measured at the MCC to calculate motor torque of the valve's motor operator. The calculated motor torque is then compared to motor torque target and limit values based on 1) packing loads, 2) thrust required to close the valve, 3) stall motor torque, and 4) valve or actuator structural limits. Entergy Report IP-RPT-04-00890, "Technical Basis for Using MCC Technology for Periodic Verification Testing at IP 2 and IP 3," states that this methodology would be used initially on MOVs with generally low safety significance and high operating margin, but also states that the report applies to all safety related MOVs at IP 2 and IP 3. Since 2004, Entergy has used the MCC methodology for periodic verification on nine safety-related MOVs: three high risk, three medium risk, and three low risk MOVs, where risk significance is defined as the combined effects of MOV risk of failure and safety significance.

Based on the available information, the team was unable to verify that the MCC method had been appropriately validated. Specifically, there did not appear to be a justified correlation between the MCC methodology calculated motor torque and actual stem thrust and torque. It was also unclear whether the MCC methodology had adequate allowances to compensate for its uncertainties in establishing MOV design basis capability (such as uncertainties related to stem friction coefficient, load sensitive behavior, and actuator efficiency) since stem thrust and stem torque are not directly measured.

MCC testing was performed in 2004 as a periodic verification test on MOV 747, the No. 21 RHR heat exchanger discharge valve, a high risk valve. The team identified that this test was invalid because this was not performed in accordance with IP-RPT-04-00890. Specifically, MOV 747 was tested using the Motor Torque Method of MCC testing which, according to IP-RPT-04-00890, is only valid for motors whose torque is between 2 and 60 foot-pounds. The motor on MOV 747 is an 80 foot-pound. motor and use of the Correlated Thrust/Torque Method was required. As a result, Entergy exceeded the six-year periodic verification test interval for MOV 747 because the last "at the valve" valid performance verification test was performed in May 2000. Entergy has provided justification for the reasonable continued operability of the valve until its scheduled testing in 2008 based on successful in-service tests, stem lubrication and actuator preventive maintenance and inspection performed in 2006. The team reviewed Entergy's basis for the operability of MOV 747 and determined that there was reasonable assurance of continued operability of the MOV.

Entergy committed to follow the Joint Owners' Group program for periodic verification of MOVs in their response to NRC Generic Letter 96-05. This periodic verification program established valve margin by measuring stem thrust and torque at the valve. Entergy's response did not indicate that the MCC method would be used for MOV periodic verification.

The MCC test method for periodic verification of MOVs is a departure from the NRC-reviewed method, which is based on direct measurement of stem thrust and torque. The acceptability of the use of the MCC methodology for periodic verification of MOVs will be an unresolved item pending further NRC review. Included with this review will be a determination of whether the MOV performance testing conducted on MOV 747 constitutes a violation of NRC requirements. **(URI 05000247/2007007-03, Use of Motor Control Center Methodology for Periodic Verification of the Design Basis Capability of Safety-Related MOVs)**

.2.1.3 Service Water Pumps and Strainers

a. Inspection Scope

The team selected the service water (SW) pumps and strainers to determine whether there was a potential for a common cause failure of the pumps and strainers. The team reviewed design documents, including drawings, calculations, procedures, the SW design basis document, tests and modifications. The team reviewed these documents to ensure the pumps and strainers were capable of meeting their design basis requirements, with consideration of allowable pump degradation, net positive suction head requirements, and strainer clogging affects. To assess the current condition of the pumps, the team interviewed engineers, and reviewed system health and related condition reports. To assess the general condition of the pumps and strainers, the team performed walkdowns of the SW pump house and strainer areas. Test results were reviewed to determine whether pump performance margins were sufficient to assure design basis assumptions could be achieved. Finally, SW system operating procedures were reviewed to ensure the system was operated in accordance with its design basis requirements.

b. Findings

No findings of significance were identified.

.2.1.4 Pressurizer Power Operated Relief Valves (PCV-455C and PCV- 456)

a. Inspection Scope

The team reviewed instrument setpoint and uncertainty calculations for pressurizer level, temperature, and pressure instruments that were relied upon for overpressure protection system operation. The team reviewed design calculations that were performed to determine the lift settings of the power operated relief valves (PORV) while in the low temperature overpressure protection (LTOP) mode of operation and related procedures to ensure the reactor coolant system (RCS) boundary is not compromised by violating the RCS pressure/temperature limits. The team also reviewed the adequacy of the backup nitrogen supply for the pressurizer PORVs, including sizing of the backup accumulator and pressure regulating setpoints, to verify the capability to cycle each PORV consistent with design basis assumptions.

b. Findings

No findings of significance were identified.

.2.1.5 Safety Injection System Check Valve (SI 847)

a. Inspection Scope

The team selected the safety injection (SI) system check valve SI 847 as representative of components whose failure posed very high risk for core damage. This valve is in the common suction line from the refueling water storage tank (RWST) for all three SI pumps. The team reviewed design drawings, vendor documents, calculations, condition reports, test procedures and results, and interviewed engineers to confirm that the valve was designed, maintained and operated in accordance with the design basis requirements. The team verified that the check valve was opening when each SI pump started, and verified, by reviewing test results, that the valve passed full flow in the open direction. The team verified that the valve operation was periodically monitored by non-intrusive testing to ensure the valve would close to prevent back-leakage.

b. Findings

No findings of significance were identified.

.2.1.6 No. 23 Emergency Diesel Generator (mechanical)

a. Inspection Scope

The team reviewed the No. 23 emergency diesel generator (EDG) to assess whether the EDG would function as required during postulated accident conditions to meet design basis requirements. The review included the fuel oil storage and supply, starting air, ventilation and combustion air, and jacket water and lube oil cooling systems. The team reviewed calculations, fuel oil transfer analyses, starting air capability analyses, heat exchanger performance analyses, system health reports, and selected condition reports to verify maintenance, testing and operation of the EDG systems were successful in meeting design basis requirements. Periodic test results and procedures were reviewed to verify fuel oil levels and transfer pump performance, starting air receiver pressures, and essential service flow rates were demonstrated and maintained within acceptable limits. The team walked down selective accessible components and areas associated with the EDG to verify proper component alignment and the absence of observed adverse material conditions that could potentially impact system operability.

b. Findings

No findings of significance were identified.

.2.1.7 No. 21 Auxiliary Feedwater Pump

a. Inspection Scope

The team reviewed the No. 21 auxiliary feedwater (AFW) pump to verify the pump was capable of achieving its design basis requirements. The review included an assessment of the condensate storage tank, procedural guidance for transfer to the alternate source of water supply for the AFW system, pump vortex protection, available net positive suction head, pump minimum flow and runout protection, and environmental and electrical qualification of equipment. The team reviewed design documents, including drawings, calculations, procedures and tests to evaluate the functional requirements of the AFW pump. Test results were reviewed to confirm that appropriate test acceptance criteria were established and that pump performance demonstrated that design basis accident assumptions would be met. Additionally, the team reviewed system health and selected corrective action reports to assess the rigor and effectiveness of corrective actions associated with Entergy's evaluation of design, maintenance, testing and operational issues. The team performed a walkdown of accessible areas of the AFW and supporting systems to verify alignment was in accordance with design basis and procedural requirements, and to assess the AFW system material condition.

b. Findings

Introduction: The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," in that, Entergy did not establish adequate design control measures to ensure the availability of the turbine driven auxiliary feedwater pump (TDAFWP) during a loss-of-offsite power (LOOP) event. Under certain LOOP situations, the team determined that the TDAFWP steam supply could be inadvertently isolated because of inadequate calculations and procedures for limiting the AFWP room temperature rise. Specifically, a calculation to determine the auxiliary feedwater pump (AFWP) room temperature rise during a LOOP did not include heat input from the TDAFWP. Further, actions that could limit the rise in AFWP room temperature and prevent the inadvertent isolation of the TDAFW pump (opening an AFWP room roll-up door or promptly restoring forced ventilation) were not included in procedures.

Description: The team reviewed calculation FCX-00086-00, "AFWP Room Temperature Rise," dated February 13, 1998. The team reviewed the calculation in part to understand the loss of ventilation effects on the TDAFWP during a LOOP. The calculation yielded relatively low temperature rises for the AFWP room, which contained two motor driven AFWPs and one TDAFWP. The highest resultant room temperature was 127.4 degrees Fahrenheit (°F), assuming an outside air temperature of 100°F. The analysis assumed the AFWP room roll-up door would be opened within 30 minutes of the onset of the LOOP. The automatic plant response to a LOOP would cause room ventilation, as well as other non-safety related loads, to be load shed from safety related electric buses. The team noted that for a LOOP event, plant procedures did not direct opening the AFWP roll-up door. The team also noted that the calculation did not appropriately consider heat input from the TDAFWP. Similar to the motor driven AFWPs, the TDAFWP would automatically start at the onset of a LOOP event.

Using the same calculation methodology, the team determined the AFWP room temperature would reach 202°F with all three AFWPs operating and the roll-up door closed. Opening the roll-up door within 30 minutes of the onset of a LOOP results in additional convection cooling, and room temperature would reach 139°F with all three AFWPs operating. However, the team considered the most limiting scenario to be the combined operation of one motor driven AFWP and the TDAFWP (assuming the other motor driven AFWP was not available or fails during the postulated event). If two motor driven AFWPs were operating, loss of the TDAFWP would be inconsequential as there would be sufficient AFW flow to support design basis assumptions.

The team noted that two temperature switches (TS) were installed at the ceiling of the AFWP room. The TSs sensed room air temperature and were designed to close the two steam isolation valves to the TDAFWP. The steam isolation valves were in series, and a single temperature switch closed the associated steam isolation valve at an established temperature of 130°F. The temperature switches were part of a design to isolate steam to the TDAFWP during a postulated high energy line break. The team noted that the steam isolation valves were air-operated and backed with a normally aligned nitrogen gas bottle such that the valves could close during LOOP conditions.

The team reviewed several recent as-found calibration data for the two AFWP room temperature switches and noted one temperature switch was found to be set at 128°F on March 5, 2005. The TS was not recalibrated and 128°F remained as the as-left setpoint.

The team reviewed setpoint data for the installed temperature switches and noted that instrument repeatability was listed at 2.25°F. The team concluded with 128°F as an allowed and actual as-left setpoint, combined with an 2.25°F instrument repeatability and other unanalyzed instrument errors, reliability of the TDAFWP was not assured if room temperature was above 125.75°F. Using the methodology of calculation FCX-00086-00, the team noted that an ambient air temperature of 93°F would result in a bulk average AFWP room temperature of 126°F with one motor driven AFWP and the TDAFWP running.

In response, Entergy entered this issue into the corrective action program for further evaluation. For the near term, Entergy implemented a standing order, performed an operability determination, and intended to proceduralize opening the roll up door or restoring forced ventilation. The team reviewed these actions and found them to be appropriate.

<u>Analysis</u>: The team determined that Entergy's failure to establish adequate design control measures to ensure the availability of the TDAFWP during a LOOP event was a

performance deficiency that was reasonably within Entergy's ability to foresee and prevent. Under certain LOOP situations, the team determined that the TDAFWP steam supply could be inadvertently isolated because of inadequate calculations and procedures for limiting the AFWP room temperature rise. Specifically, a calculation to determine the auxiliary feedwater pump (AFWP) room temperature rise during a LOOP did not include heat input from the TDAFWP. Further, actions that could limit the rise in AFWP room temperature and prevent the inadvertent isolation of the TDAFW pump (opening an AFWP room roll-up door or promptly restoring forced ventilation) were not included in procedures.

This issue was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the team conducted a Phase 1 SDP screening and determined the finding was of very low safety significance (Green) because it did not represent the loss of safety function of the TDAFWP (single train) for greater than its 72 hour technical specification allowed outage time, based on the team's determination that over the last year, the site ambient temperature was not above 93°F for more than 72 hours. The team reviewed site weather data for calendar year 2006, and noted that ambient air temperature greater than 93°F existed on five days, but based on the mean temperature on those days, the temperature at the site was not above 93°F for more than 72 hours.

Enforcement: 10 CFR 50 Appendix B, "Design Control," requires, in part, that measures shall be established to assure that the design basis for structures, systems, and components are correctly translated into procedures. Contrary to the above, as of February 13, 1998, calculation FCX-00086-00, "AFWP Temperature Rise," did not appropriately analyze environmental effects on the operability and availability of the TDAFWP, and actions to promptly open a roll-up door to the AFWP room were not incorporated into LOOP procedures. Because this violation is of very low safety significance and has been entered into Entergy's correction action program (CR-IP2-2007-00656, 00659, and 00662), this violation is being treated as a non-cited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000247/2007007-04, Inadequate Design Control for Environmental Effects to Ensure the Availability of the Turbine Driven Auxiliary Feedwater Pump Operation)**

.2.1.8 No. 23 Charging Pump

a. Inspection Scope

The team reviewed the No. 23 charging pump to verify its capability to meet design basis assumptions with respect to pump flow and pressure. The team reviewed calculations, drawings, procedures, tests, and other analyses to verify selected calculation inputs, assumptions, and methodologies were accurate and justified, and were consistently applied. The available net positive suction head for the charging pump and the cooling water flow rates for the pump drive unit and frame lube oil coolers were verified to be consistent with design assumptions to ensure reliable pump operation. The team reviewed completed tests to confirm the acceptance criteria and test results demonstrated the capability of the pump to provide required flow rates. The team reviewed system health and selective corrective action reports to assess the identification and disposition of maintenance, testing, and operational issues. The team also conducted a walkdown of accessible components and features associated with the charging pump to verify the material condition of the pump and support systems features would not adversely affect system performance.

b. Findings

No findings of significance were identified.

.2.1.9 Auxiliary Feedwater System Check Valves (BFD-31, -34, and -39)

a. Inspection Scope

The team inspected AFW system pump discharge check valves BFD-31, -34, and -39 to verify that each valve can open to pass the required design basis AFW system forward flow and can close to prevent reverse flow. The inspection included a review of periodic test results, verification of the bases for flow acceptance criteria, and documentation of demonstration of closure to prevent reverse flow. The tests and criteria were demonstrated in a combination of quarterly recirculation flow and refueling outage full flow tests. The team reviewed non-intrusive and open-and-inspect test results to confirm satisfactory check valve performance. The team also reviewed the corrective action program, work control system database and system health reports to assess whether there were any adverse maintenance or performance trends with these valves.

b. Findings

No findings of significance were identified.

.2.1.10 Gas Turbine No. 1

a. Inspection Scope

The team conducted interviews with engineers, conducted a walkdown of the equipment, and observed operation of gas turbine 1 (GT-1). The team also reviewed GT reliability and unavailability records, operator logs, condition reports, procedures, completed surveillances, modifications, the GT system reliability action plan, and maintenance rule basis documents to verify the reliability and capability of GT-1 to provide an alternate alternating current (AC) power source for station blackout and 10 CFR 50, Appendix R fire scenarios. In particular, the team reviewed the capability of GT-1 to perform a “black start” without any AC power available. The team reviewed 20 volts direct current (Vdc) and 125Vdc battery sizing data and vendor data to evaluate the ability of the gas turbine support systems (black start diesel, starting diesel, and other auxiliaries) to perform their functions without AC power available.

b. Findings

1. (Closed) URI 05000247/2006005-03: Reliability / Unavailability of the Gas Turbine System and Impact on Functionality

Introduction: The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50.65(a)(1), the Maintenance Rule, in that, Entergy failed to monitor the GT system in a manner that provided reasonable assurance that the system could perform its intended safety function. Specifically, Entergy did not establish appropriate GT reliability goals, and therefore did not take corrective actions, when GT-1 had exceeded these goals for maintenance preventable functions failures (MPFF). In addition, Entergy did not properly classify repeat MPFFs that resulted in a similar failure to take corrective actions as required. This resulted in additional GT-1 out of service time that would not have happened if appropriate actions had been taken.

Description: Gas turbines 1 and 3 (GT-1 and GT-3), are credited in Entergy's analysis to cope with station blackout and 10 CFR 50, Appendix R fire scenarios to ensure safe shutdown of the reactor. The system, consisting of GT-1, GT-3 and associated support systems, is classified as risk-significant in accordance with Entergy's Maintenance Rule program. This system has been in a category (a)(1) monitoring status since the inception of the Maintenance Rule in 1996 due to the system's failure to achieve the established reliability goals, availability goals, or both during the last ten years. An (a)(1) action plan had been established to improve overall system performance.

The team reviewed the applicable Maintenance Rule basis document and evaluated the established monitoring goals for availability and reliability. Specific to reliability, the established goal was less than or equal to five MPFFs and no repeat maintenance preventable functional failures (RMPFFs) in a 24 month rolling cycle. The number of allowable MPFFs was calculated under the assumption that there would be, on average, 82 start demands during the 24 month cycle. The team reviewed the operating history over the last three years and determined that the number of start demands averaged 38 (per 24 month cycle). The review showed that the assumption for the number of start demands was calculated based on data from a period when the GTs were routinely run to provide peaking power. Entergy stopped using the turbines for this purpose in 2000; however, they did not account for this change in operation from when the Maintenance Rule goal was evaluated and established. Based on this calculation methodology, the team determined that the appropriate goal for MPFFs should be less than or equal to two rather than five. GT-1 currently has three MPFFs over the last 24 month period, the last of which occurred on August 6, 2006.

The team also reviewed work orders and condition reports associated with the GTs for the last two years. During this review, the team noted four failures of GT-1 due to low coolant level in the starting diesel heat exchanger. Entergy determined that the first of the four failures was due to a design deficiency associated with the GT-1 starting diesel coolant system; and Entergy classified this failure as a MPFF. The three subsequent failures were all classified as design deficiencies but none of them were similarly classified as MPFFs. The team reviewed the details associated with each failure and the guidance in NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," and determined that two of the remaining three failures should have been classified as RMPFFs because they were due to the same design deficiency. The other failure involved additional failure mechanisms that would not have resulted in a MPFF or RMPFF classification. As a result of the two failures that should have been RMPFFs, the established goal of no RMPFFs was exceeded (on August 6, 2006).

The team determined that had the reliability goal been appropriate and justifiable, or the functional failures been appropriately classified, a review of the current (a)(1) action plan and its associated corrective actions to improve system performance would have been required. This would have resulted in additional corrective actions being performed to improve system performance in order to meet the required reliability goal. This review was not done, and no additional actions were implemented to improve GT availability and reliability. As a result, Entergy incurred an additional GT-1 RMPFF that could have been avoided had effective corrective actions been taken to meet reliability goals.

Entergy entered this issue into their corrective action program. Entergy's short term corrective actions included lowering the allowable goal for MPFFs to less than or equal to two; and revising the GT-1 (a)(1) action plan to improve reliability. The team found these corrective actions to be appropriate.

Analysis: The team determined Entergy's failure to monitor the GT system in a manner that provided reasonable assurance that the system could perform its intended safety function was a performance deficiency that was reasonably within Entergy's ability to foresee and prevent. Entergy did not establish appropriate GT reliability goals, and therefore did not take corrective actions, when GT-1 had exceeded these goals for MPFFs. Specifically, Entergy did not update its reliability goals to reflect the reduction in start demands of the GTs and did not recognize that GT-1 had exceeded its allowed MPFFs. In addition, Entergy did not properly classify RMPFFs that resulted in a similar failure to take corrective actions as required. Consequently, Entergy did not update the Maintenance Rule (a)(1) action plan to improve GT reliability and availability. Further, this resulted in additional GT-1 out of service time that would not have happened if appropriate actions had been taken.

The finding was more than minor because it was similar to NRC Inspection Manual Chapter 0612, Appendix E, "Examples of Minor Issues," Example 7.a, in that, appropriate GT reliability goals were not commensurate with safety, and appropriate corrective actions were not taken when goals were not met. This finding was associated with the equipment performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The GTs are credited as an alternate AC power source for both station blackout and 10 CFR 50, Appendix R fire scenarios. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements.

In accordance with NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the team conducted a Phase 1 SDP screening and determined a more detailed Phase 2 SDP evaluation was required to assess the safety significance because the finding represented an actual loss of safety function of a non-Technical Specification required train of equipment designated as risk-significant per 10 CFR 50.65 for greater than 24 hours. The team used the Risk-Informed Inspection Notebook for Indian Point Nuclear Generating Station Unit 2, to conduct the Phase 2 evaluation. Because of a lack of detail (only dates provided, not hours) in the data reviewed in condition reports and GT operating data, the team could not determine the exact number of additional hours that GT-1 was unavailable over the one-year period ending September 30, 2006, related to the two RMPFFs. However, by reviewing the associated dates in the data, the team was able to approximate that the additional time was between two and three days. Accordingly, the team applied an initiating events likelihood of less than three days. The Phase 2 approximation yielded a result of very low safety significance (Green). The most dominant accident sequence involved a loss-of-offsite power, and the subsequent failure of two emergency diesel generators in addition to the failure to restore power within five hours via the off-site power network or one of the gas turbines [LOOP (4) + EAC (3) + REC5 (2) = 9].

The finding has a cross-cutting aspect in the area of human performance because Entergy did not adequately ensure procedures were complete, accurate, and up-to-date. Specifically, procedure ENN-DC-171, "Maintenance Rule Monitoring," did not provide steps to discriminate between the classification of an initial design deficiency and further failures due to the same condition, resulting in mis-classifying several GT functional failures.

<u>Enforcement</u>: 10 CFR 50.65(a)(1) requires, in part, that holders of an operating license shall monitor the performance or condition of structures, systems, and components (SSC) within the scope of the rule as defined by 10 CFR 50.65(b), against licensee established goals, in a manner sufficient to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions; and when the performance or condition of a SCC does not meet established goals, appropriate corrective actions shall be taken. Contrary to the above, as of August 6, 2006, Entergy failed to monitor the condition of the GT system in a manner to provide reasonable assurance the system could perform its intended function. The established goals for reliability were not justified and Entergy failed to properly evaluate RMPFFs. These errors resulted in Entergy not evaluating and taking appropriate corrective actions to improve system performance. Because this violation is of very low safety significance and has been entered in Entergy's corrective action program (CR-IP2-2006-06842), this violation is

being treated as a non-cited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000247/2007007-05, Failure to Adequately Monitor Gas Turbine System Performance as Required by the Maintenance Rule)**

2. Failure to Correct Degraded Gas Turbine 1 Reliability

Introduction: The team identified a finding of very low safety significance (Green) involving Entergy procedure, EN-LI-102, "Corrective Action Process," in that, Entergy failed to take corrective actions to address degraded GT-1 reliability. This resulted in a two and one half day time period in January 2007 when GT-1 and GT-3 were simultaneously inoperable because, after GT-3 was made inoperable for planned maintenance activities, GT-1 was subsequently found to be inoperable. Specifically, the reliability of GT-1 declined from an average of 75% for 2005 and the first 10 months of 2006, to 50% for the three months from November 2006 to January 2007; however, Entergy did not take actions to correct this degraded reliability.

Description: Gas turbines 1 and 3 (GT-1 and GT-3) are credited in Entergy's analysis to cope with station blackout and 10 CFR 50, Appendix R fire scenarios to ensure safe shutdown of the reactor. UFSAR Section 8.2.1.1, "Reliability Assurance," states that at least one gas turbine generator and associated switchgear and breakers shall be operable at all times. If both GTs are inoperable, the UFSAR requires the system be restored to operable within seven days or a plant shutdown be performed.

From January 2005 to about October 2006, GT-1 reliability, as measured by the ability of the gas turbine to start and load during surveillances, averaged 75%. However, during the months November 2006 to January 2007, GT-1 reliability declined to 50%. In particular, GT-1 failed each month's surveillance due to multiple automatic trips, and the surveillance had to be re-performed on the following day. On two of the subsequent successful surveillances, multiple starts were required to ultimately achieve a satisfactory surveillance result.

After a successful surveillance of GT-1 on January 16, 2007, the NRC senior resident inspector questioned Entergy personnel whether the successful surveillance was sufficient to establish GT-1 operability given the recent poor GT-1 reliability. In response, Entergy initiated condition report CR-IP2-2007-00259, and completed a Basis for Functionality assessment in accordance with Procedure EN-OP-104, "Operability Determinations," which concluded that GT-1 was operable (functional). The assessment recommended that additional GT-1 testing should be conducted during the week of January 15 and additional dates (accelerated testing) in accordance with 2-SOP-31.1.2, "GT-1 Local Operations," to further establish GT-1 reliability. The Basis for Functionality assessment's conclusion was based on the successful GT-1 surveillance on January 16, 2007, as well as previously completed corrective actions. The team determined that the functionality assessment was inadequate because, although it recognized and documented an adverse GT-1 reliability trend, the large number of recent automatic shutdowns (10) and equipment failures (3) were not collectively evaluated and considered for overall GT system reliability and functionality. The team concluded that had Entergy performed an adequate analysis, they should have concluded that GT-1 was not reliable, and therefore, non-functional.

Following the GT-1 surveillance on January 16, at 12:14 p.m., a 13.8 kV bus section (13W94) was removed from service for switchyard maintenance. This action rendered GT-3 inoperable due to the unavailability of the associated switchgear and breakers. On January 18, while the switchgear remained out-of-service, Entergy performed a surveillance on the GT-1 in accordance with the recommendations from Entergy's Basis for Functionality document. The equipment failed the surveillance, resulting in GT-1 being declared inoperable. Bus section 13W94 was returned to service on January 18 at 11:45 p.m., thereby restoring GT-3 to an operable status. The team concluded that both GT-1 and GT-3 were unavailable during the time period January 16 - 18, 2007. Entergy entered this issue into their corrective action program, and developed an action plan to address GT reliability issues. In addition, GT-1 remained out of service to investigate and repair the reliability challenges. The team reviewed Entergy's GT system action plan, and found it to be adequate.

Analysis: The team determined Entergy's failure to take corrective actions to address degraded GT-1 reliability, in accordance with Entergy procedure, EN-LI-102, "Corrective Action Process," was a performance deficiency that was reasonably within Entergy's ability to foresee and prevent. The procedure stated that actions for a condition report should be determined, implemented and adequate to resolve the condition. Although Entergy initiated a condition report and completed a functionality assessment, the associated evaluation and corrective actions were not adequate. This resulted in a two and one half day time period in January 2007 when GT-1 and GT-3 were simultaneously inoperable because, after GT-3 was made inoperable for planned maintenance activities, GT-1 was subsequently found to be inoperable. Specifically, the reliability of GT-1 declined from an average of 75% for 2005 and the first 10 months of 2006, to 50% for the three months from November 2006 to January 2007; however, Entergy did not correct this degraded reliability.

The issue was more than minor because it was associated with the equipment reliability attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The GTs are credited as the alternate AC power source for both station blackout and 10 CFR 50, Appendix R fire scenarios, as stated in UFSAR Section 8.2.1.1.

In accordance with NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the team conducted a Phase 1 SDP screening and determined a more detailed Phase 2 SDP evaluation was required to assess the safety significance because the finding represented an actual loss of safety function of a non-Technical Specification required train of equipment designated as risk-significant per 10 CFR 50.65 for greater than 24 hours (GT-1 and GT-3 were simultaneously unavailable for two and one half days from January 16 - 18, 2007). The team used the Risk-Informed Inspection Notebook for Indian Point Nuclear Generating Station Unit 2, to conduct the Phase 2 evaluation using an initiating events likelihood of less than three days with an assumed inability to recover AC power from the GTs in five hours (normal offsite power could still have been recovered). The most dominant accident sequence involved a station blackout (loss-of-offsite power with the failure of two emergency diesel generators) and failure to recover offsite power in five hours [LOOP (4) + EAC (3) + REC5 (1) = 8]. The Phase 2 evaluation yielded a result of very low safety significance (Green).

The finding has a cross-cutting aspect in the area of problem identification and resolution because Entergy did not correct the degraded reliability of GT-1, resulting in having GT-1 and GT-3 simultaneously inoperable.

Enforcement: No violation of regulatory requirements occurred. The team determined that the finding did not represent a non-compliance because the gas turbine system is not a safety related system. Entergy entered this issue in their corrective action program (CR-IP2-2007-00259 and CR-IP2-2007-00308), and developed a revised GT system reliability action plan. **(FIN 05000247/2007007-06, Failure to Correct Degraded Gas Turbine 1 Reliability)**

.2.1.11 Residual Heat Removal Sump Isolation Valve (MOV 885A)

a. Inspection Scope

The team conducted interviews with engineers and reviewed calculations, procedures, periodic verification test results, and technical reports to verify the capability of MOV 885A to perform its intended function during postulated design basis accident conditions. NRC Generic Letter 96-05 (related to MOV periodic verification) correspondence was reviewed to verify that Entergy was meeting its commitments for periodic verification of the valve. Preventive maintenance requirements and corrective action reports were also reviewed in order to determine the performance and operational history of the valve.

b. Findings

No findings of significance were identified.

.2.1.12 Flooding in the 480Vac Switchgear Room

a. Inspection Scope

The team conducted a walkdown of the 480Vac switchgear room and reviewed the IP 2 Individual Plant Examination of External Events, Probabilistic Safety Analysis Flooding Analysis, drawings, and related evaluations to verify the conclusions made in the analyses with respect to potential flooding scenarios were accurate and conservative. The team interviewed engineers and reviewed calculations, operating procedures, condition reports, and fire protection and service water inspection results to verify that Entergy had taken steps to minimize the chance of flooding and had procedures in place to minimize the consequences of flooding.

b. Findings

No findings of significance were identified.

.2.1.13 Station Battery No. 21

a. Inspection Scope

The team reviewed the No. 21 battery design calculations to verify that the battery sizing would satisfy the requirements of the safety related and risk significant DC loads, and that the minimum possible voltage was taken into account. In particular, the evaluation focused on verifying that the battery was adequately sized to supply the design duty cycle of the 125Vdc system for the loss-of-coolant accident/loss-of-offsite power (LOCA/LOOP) and Station Blackout loading scenarios, and that adequate voltage would remain available for the individual load devices required to operate during the scenario durations. Plant drawings were reviewed to ensure that all loads were considered. The No. 21 battery charger sizing calculation was reviewed to evaluate whether it was consistent with the design and licensing bases. The team reviewed the DC protective coordination study to verify that breaker and fuse coordination was provided for postulated faults in the DC system.

In addition, a walkdown was performed to evaluate the condition of the battery and battery charger. The team reviewed battery test procedures and results to determine whether test acceptance criteria and frequency requirements specified in technical specifications and appropriate standards were satisfied. Engineers were interviewed regarding design aspects and operating history for the battery, and a sample of condition reports was selected to verify that design and testing issues related to the No. 21 battery were adequately addressed.

b. Findings

1. Inadequate Station Battery Capacity Testing for Degradation Monitoring

Introduction: The team identified a finding of very low safety significance (Green) involving a non-cited violation of Technical Specification 3.8.6.6, in that, Entergy did not perform station battery capacity testing in accordance with IEEE Standard 450-1995 (related to battery maintenance and testing). Specifically, Entergy procedurally terminated battery capacity testing at the rated discharge time (four hours), before reaching the minimum voltage, as specified by IEEE Standard 450-1995. This prevented accurate quantitative measurement of capacity degradation and identification of the need to conduct potential accelerated battery testing, as specified by both IEEE Standard 450-1995 and the technical specifications, if battery capacity drops by more than 10% relative to the previous test.

Description: The team reviewed the performance test results for battery No. 21 to ensure adequate capacity was available and to verify that testing is performed in accordance with IEEE Standard 450-1995 and the technical specifications.

Performance testing is required every five years by Technical Specification 3.8.6.6 to verify adequate capacity and performance of the battery, but done every two years at IP 2.

The team noted that according to step 7.2.4 of the performance test procedure, PT-R76A, “Station Battery 21 Load,” the test duration is set to four hours. Since four hours is the rated time for the discharge rate used, this typically terminates the test prior to reaching battery minimum voltage.

The team found that Technical Specification 3.8.6.6 requires the capacity to be normally measured every 60 months, but the testing frequency is to be increased to every 12 months when the battery shows degradation. According to the technical specifications and IEEE Standard 450-1995, degradation is indicated when the battery capacity drops by more than 10% relative to its capacity on the previous performance test. This is the measured battery capacity at its minimum voltage (fully discharged), and compared with the prior performance test.

Battery capacity typically peaks between 110% and 115% and capacity is greater than 100% for most of the life of the battery. By ending the capacity tests prior to reaching battery minimum voltage, quantitative measurement of degradation is only possible if the battery has had two tests below 100% capacity (the design basis is that they are operable if over 80% capacity). Therefore, for most battery tests, Entergy has been unable to quantitatively evaluate the technical specification testing frequency based on the potential 10% degradation criteria that is measured at battery minimum voltage.

Entergy subsequently analyzed the previous test results to calculate the capacity changes for possible degradation in station batteries Nos. 22 and 24. Station battery No. 21 increased in capacity as is normal early in battery life, and station battery No. 23 was below 100% capacity for two test cycles, so its capacity was directly measured (and no degradation was apparent). The largest calculated capacity decrease was 7% in two years for the No. 24 battery, which was less than the limit of 10%. However, provided the calculated value was accurate and extrapolating the calculated results to a five year testing interval as is allowed by technical specifications, then the capacity could decrease significantly. The team found reasonable assurance of operability for the station batteries after considering that the margin of error for the calculated values of degradation due to the non-linear characteristics of the battery discharge is large; the battery capacities for batteries Nos. 22 and 24 were measured to be above 100% with an acceptance criteria of 80%; and the weekly, monthly and quarterly test results for the batteries have been satisfactory. Entergy entered the issue into their corrective action program (CR-IP2-2007-00193), and intended to appropriately revise the testing procedures.

Analysis: The team determined that Entergy’s failure to perform station battery capacity testing in accordance with IEEE Standard 450-1995 (related to battery maintenance and testing), as required by Technical Specification 3.8.6.6, was a performance deficiency that was reasonably within Entergy’s ability to foresee and prevent. Specifically, Entergy procedurally terminated battery capacity testing at the rated discharge time (four hours), before reaching the minimum voltage, as specified by IEEE Standard 450-1995. This prevented accurate quantitative measurement of capacity degradation and identification of the need to conduct potential accelerated battery testing, as specified by both IEEE Standard 450-1995 and the technical specifications, if battery capacity drops by more than 10% relative to the previous test.

This issue was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the team conducted a Phase 1 SDP screening and determined the finding was of very low safety significance (Green) because it did not result in a loss of safety system function, based upon 1) the team's verification of Entergy's calculations, which showed less than 10% decrease in the station battery capacities, and 2) other battery testing.

Entergy determined that the battery test procedure was inadequate, and the 2003 Improved Technical Specification project represented a missed opportunity to identify the procedure error, although the procedure error apparently existed long before the 2003 project. The team determined that because 1) the 2003 missed opportunity was not a significant contributor to the cause of the finding, and 2) the error was not reflective of current performance, there was not a cross-cutting aspect to this finding.

Enforcement: Technical Specification 3.8.6.6 requires in part that battery capacity testing be done at an increased frequency when the battery shows degradation. The basis for Technical Specification 3.8.6.6 states that, "degradation is indicated, according to IEEE Standard 450-1995, when the battery capacity drops by more than 10% relative to its capacity on the previous performance test," and to "maintain the discharge rate until the battery terminal voltage decreases to a value equal to the minimum average voltage per cell . . . times the number of cells." Contrary to the above, as of January 12, 2007, testing was terminated prior to reaching battery minimum voltage, which prevented measuring the capacity quantitatively above 100%. Because this violation is of very low safety significance and has been entered into Entergy's corrective action program (CR-IP2-2007-00193), this violation is being treated as a non-cited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000247/2007007-07, Inadequate Station Battery Capacity Testing for Degradation Monitoring)**

2. Ineffective Corrective Action for High Inter-Tier Battery Resistances

Introduction: The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective

Action," in that, Entergy did not take effective corrective actions for out-of-tolerance inter-tier resistances on the No. 21 station battery. Specifically, after repeated failures of the station battery No. 21 inter-tier resistance testing, vendor and IEEE Standard 450-1995 recommended corrective actions were not taken to correct the adverse out-of-tolerance resistance trend.

Description: The team reviewed past results from the annual performance test 2-PT-A035A, "21 Station Battery Inter-Cell Resistance Checks," to verify that inter-cell and inter-tier resistance checks were performed properly and any out-of-tolerance measurements were addressed.

The team noted that various inter-tier connections had failed the resistance test for each of the last four years. Of particular note was the inter-tier connection between cells 40 and 41, which had failed all four years. The dominant concern for high resistance connections in the battery is that when loaded during accident conditions, they can cause localized heating that can cause melting of the connection or other permanent damage to the battery. The corrective action for high resistance connections, based on both IEEE Standard 450-1995 and the battery vendor manual, is to retorque the connections and retest. If this does not correct the out-of-tolerance condition, then the connection should be disconnected, cleaned, and remade.

After the test failures (for high inter-tier resistance) in March 2004 and February 2005, actions were taken to address re-baselining affected connections due to a change in the resistance measurement instrument, but the out-of-tolerance connections were not retorqued or cleaned. In January 2006, the failed connections were retorqued. The out-of-tolerance inter-tier connections for the 2004, 2005, and January 2006 tests were only slightly greater than the acceptance criteria, but were generally trending higher. When the most recent test was completed on December 4, 2006, two inter-tier connections were significantly out-of-tolerance, and the remaining inter-tier connections were indeterminate since portions of the inter-tier cables had been removed without re-baselining the data. The acceptance criteria is 20% above baseline, and the worst connection was approximately 43% above the baseline. A work order was written to retorque the connections; however, there was no retest or additional actions taken for the connections.

In response to this item, Entergy entered the issue into their corrective action program (CR-IP2-2007-00737) and issued a work order to retest the connections and take appropriate actions based on the results. In addition, Entergy performed calculations, which demonstrated that the voltage drop due to the as-found resistance of the inter-tier connections was small and did not impact No. 21 battery operability. The team verified that these calculations demonstrated No. 21 station battery operability, and that Entergy's completed and planned corrective actions were appropriate.

Analysis: The team determined that Entergy's failure to take effective corrective actions for a condition adverse to quality concerning out-of-tolerance inter-tier resistances on the No. 21 station battery was a performance deficiency that was reasonably within

Entergy's ability to foresee and prevent. Specifically, after repeated failures of the No. 21 station battery inter-tier resistance testing, vendor and IEEE Standard 450-1995 recommended corrective actions were not taken to correct the adverse out-of-tolerance resistance trend.

This issue was more than minor because if it was left uncorrected, it would have become a more significant safety concern. Specifically, high resistance connections in a battery that is loaded during accident conditions can cause localized heating and can cause melting of the connection or other permanent damage to the battery. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the team conducted a Phase 1 SDP screening and determined the finding was of very low safety significance (Green) because it did not result in a loss of safety system function, based upon the team's verification of Entergy's revised calculations, which demonstrated that the voltage drop due to the as-found resistance of the inter-tier connections was small and did not impact No. 21 battery operability.

The finding has a cross-cutting aspect in the area of problem identification and resolution because Entergy did not implement timely and effective corrective actions to address an adverse trend of out-of-tolerance battery inter-tier resistances.

Enforcement: 10 CFR 50 Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to the above, between March 4, 2004, and February 9, 2007, measures had not been established to ensure that the recommendations of the battery vendor and IEEE Standard 450-1995 were implemented when an adverse trend of out-of-tolerance battery inter-tier resistances were discovered during the last four annual resistance checks for No. 21 station battery. Because this violation is of very low safety significance and has been entered into Entergy's corrective action program (CR-IP2-2007-00737), this violation is being treated as a non-cited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000247/2007007-08, Ineffective Corrective Action for High Inter-Tier Battery Resistances)**

3. Untimely Corrective Actions for Decrease in Battery Margin

Introduction: The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," in that, Entergy did not promptly identify and correct a condition adverse to quality. Specifically, Entergy did not recognize at the appropriate time the need to write a condition report, perform an operability determination, or place controls on the use of the No. 23 battery design calculations when errors were discovered in the No. 23 battery design calculations that significantly lowered the battery capacity margin.

Description: During a review of battery sizing and voltage drop calculations, the team noted several errors that reduced the capacity margin for various batteries. In an effort to show that these errors were previously identified and were being incorporated into the newest revision of the calculations, the team was offered the draft revisions to the calculations. The draft sizing (capacity) and voltage drop calculations were of generally high quality.

Entergy had recognized during the design review process of the new calculations that some loads were inadvertently omitted in the draft versions, such that corrections needed to be made prior to issuing the calculations. The effect of the new loads would be to lower the final capacity margin. Although the margin generally appears to have decreased for all station batteries, it was apparent that there will be sufficient margin for the Nos. 21, 22, and 24 batteries when the calculations are complete. But, after qualitatively considering the effects of the changes based on the approximate magnitude of the loads to be added and the stated capacity margin in the new calculations, the team questioned the margin, and potentially the operability, of the No. 23 battery.

Entergy's corrective actions included performing an operability review for this issue. In particular, Entergy performed a calculation, and acknowledged a potential operability concern at low room temperatures (< 65°F). The team performed an independent calculation, using Entergy's calculation inputs, data, and methodology. When corrected for current battery age and typical minimum temperature, the result of the team's calculation yielded about a negative 3% margin. In response, Entergy identified six conservatisms in the draft calculation that can be refined to restore margin. The team qualitatively reviewed the conservatisms and agreed that there was a reasonable basis for current operability.

Notwithstanding, the team performed independent calculations to consider future operability and found that without making any changes to the conservative assumptions, the design capacity (at the end of the life of the battery and at the lowest design temperature) would be negative by 17%. The calculation of record (corrected for lowest design temperature) showed a positive capacity margin of 22%. Entergy had previous plans to replace the No. 23 battery in the next outage, which will prevent the decline in margin due to age degradation. The team found reasonable assurance of operability through the next outage (when the battery will be replaced), but it was unclear whether operability would have been assured for the design life of the battery.

Entergy received the draft calculations from a contractor in April 2006, but due to formatting issues, did not receive the final draft copy until September 2006. Therefore, Entergy had from September 2006 until the team raised the issue in February 2007 to use the corrective action process to formally document and confirm operability and prevent the inaccurate calculations of record from being used without considering the new information about decreased battery capacity. Because the new draft calculations showed a significant margin decrease that potentially affected current operability, and because the new draft calculations showed an even greater margin decrease at design conditions, it was not appropriate to delay initiating measures to ensure operability, such as writing a condition report, performing an operability determination, or placing controls on the use of the No. 23 battery sizing and voltage drop calculations of record. As a result of the inspection, Entergy entered the issue into their corrective action program (CR-IP2-2007-00842).

Analysis: The team determined that Entergy's failure to promptly identify and correct a condition adverse to quality, regarding known errors in the No. 23 station battery design calculations, was a performance deficiency that was reasonably within Entergy's ability to foresee and prevent. Specifically, actions were not taken to write a condition report, perform an operability determination, or place controls on the use of the No. 23 battery design calculations when errors were discovered in the No. 23 battery design calculations that significantly lowered the battery capacity margin.

This issue was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the team conducted a Phase 1 SDP screening and determined the finding was of very low safety significance (Green) because it did not result in a loss of safety system function, based upon the team's verification of Entergy's revised calculations, which demonstrated operability through the next refueling outage.

The finding has a cross-cutting aspect in the area of problem identification and resolution because Entergy failed to promptly identify the decrease in margin found in the No. 23 battery design calculations of record.

Enforcement: 10 CFR 50 Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to the above, between September 2006 and February 15, 2007, measures had not been established to ensure that the decrease in design margin associated with the No. 23 battery was promptly identified and corrected. Because this violation is of very low safety significance and has been entered into Entergy's corrective action program (CR-IP2-2007-00842), this violation is being treated as a non-cited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000247/2007007-09, Untimely Corrective Actions for Decrease in Battery Margin)**

.2.1.14 138kV Switchyard

a. Inspection Scope

The team reviewed the capability of the 138kV Switchyard to provide offsite power to IP 2. The team walked down the switchyard to observe the material condition of the general area, breakers, switches, protective relaying equipment, battery backup power, and power lines. The responsible engineers were interviewed regarding the coordination between Entergy and ConEd for maintenance and operation of the switchyard, the historical and recent maintenance issues with switchyard equipment, and upgrades to the switchyard. The Maintenance Rule basis document and associated action plan for the 138kV system were reviewed for completeness and effectiveness of managing unavailability and unreliability of the switchyard. The system health report was reviewed to evaluate the managing of system challenges and future plans.

Maintenance history was reviewed for indicating and protective equipment (lightning arresters, current transformers, and potential transformers) to verify that Entergy was adequately testing and maintaining the equipment. Testing performance and vendor information were reviewed for the station auxiliary transformer and breaker BT4-5, which are risk significant portions of the 138kV system, to verify that the equipment was being tested and operated in accordance with the vendor guidance. A sample of condition reports was selected to verify that issues related to the 138kV switchyard were adequately addressed.

b. Findings

No findings of significance were identified.

.2.1.15 Instrument Bus No. 23

a. Inspection Scope

The team reviewed the capability of the No. 23 instrument bus to provide instrumentation and control power during all conditions, but particularly during design basis accident conditions. The calculation for loading and voltage drop was reviewed for the No. 23 instrument bus to ensure that sufficient capacity exists for all normal and accident loading, and that sufficient voltage was available for all loads. The No. 23 battery loading study was reviewed to verify that the battery was capable of providing the appropriate load for the inverter. The team interviewed engineers to determine past and current issues related to the system. The system health report was reviewed to evaluate the management of system challenges. A sample of condition reports was selected to verify that issues related to the No. 23 instrument bus were adequately addressed.

b. Findings

No findings of significance were identified.

.2.1.16 480Vac Switchgear - Bus 6A and Breaker 6A (Station Service Transformer Breaker)

a. Inspection Scope

The team reviewed condition reports and corrective and preventive maintenance procedures for Bus 6A and Breaker 6A to evaluate the reliability of equipment. The team reviewed the electrical distribution system load flow analysis and the calculation that evaluated the overload capability for the Westinghouse type DB-75 circuit breaker and 480Vac switchgear to determine the operating margin for components that were identified by calculation as limiting components during overload conditions. The team conducted walkdowns of the switchgear to observe the material condition and operating environment for indications of degradation of equipment. The team reviewed drawings, calculations, data sheets, and calibration tests to determine whether breaker 6A overcurrent trip settings were appropriately selected and tested in accordance with the established acceptance criteria. The team also reviewed the degraded voltage relay setpoint and uncertainty calculations, calibration test acceptance criteria, and relay calibration test results to determine whether the degraded voltage relay settings were in accordance with technical specification requirements.

b. Findings

No findings of significance were identified.

.2.1.17 Emergency Diesel Generator Fuel Oil Transfer Pump Motor and Starter

a. Inspection Scope

The team reviewed fuel oil storage and transfer system design basis documents, including drawings, procedures, calculations and modifications to determine whether the pump, valve controls and alarm functions were in accordance with system design basis requirements. The team reviewed condition reports, corrective and preventive maintenance, and testing for the fuel oil transfer pump motors, motor starters, and fuel oil day and storage tank level switches and alarms to determine the reliability of equipment. The team reviewed the fuel oil transfer pump motor starter breaker and motor feeder cable sizing, and the motors thermal overload protection to determine whether the components were sized in accordance with design basis requirements. The team also conducted walkdowns of the components to assess the material condition and operating environment of the equipment, and to determine that the installed components associated with the No. 23 EDG fuel oil transfer pump motor starter were in accordance with design analyses.

b. Findings

No findings of significance were identified.

.2.1.18 Emergency Diesel Generator No. 23 (electrical), Starting Circuit, and Output Breaker

a. Inspection Scope

The team reviewed the EDG drawings and the schematics for the starting circuit, vendor data for the NST time delay relay and the diesel starting air motor solenoid, and calculations to determine whether the selected components were maintained and operated in accordance with the design bases. The team reviewed the EDG loading study for the design basis loading conditions to determine the operating margin available on the EDG and generator breaker ratings. The team conducted walkdowns of the EDGs and associated support equipment to determine the material condition and operating environment for indications of degradation of equipment.

b. Findings

No findings of significance were identified.

.2.1.19 RHR Heat Exchanger Discharge Valves MOV-746 and MOV-747 (Motors and Starters)

a. Inspection Scope

The team reviewed the one-line diagrams and schematics and the electrical distribution system load flow and valve motor circuit voltage analyses to determine the minimum voltage available at the valve motor terminals during degraded voltage conditions. The team also reviewed the valve motor thermal overload sizing calculation to verify that the selected heater was considered in the minimum voltage analysis. The team reviewed the valve operator thrust analysis to verify that the minimum voltage available was considered when calculating the available thrust margin. The team also reviewed condition reports and the corrective maintenance history for the valve motors and motor starters to determine the reliability of equipment.

b. Findings

No findings of significance were identified.

.2.2 Detailed Operator Action Reviews (5 Samples)

The team assessed manual operator actions and selected a sample of five operator actions for detailed review based upon risk significance, time urgency, and factors affecting the likelihood of human error. The operator actions were selected from a PRA ranking of operator action importance based on RAW and RRW values. The non-PRA considerations in the selection process included the following factors:

- Margin between the time needed to complete the actions and the time available prior to adverse reactor consequences;
- Complexity of the actions;
- Reliability and/or redundancy of components associated with the actions;
- Extent of actions to be performed outside of the control room;
- Procedural guidance; and
- Training.

.2.2.1 AC Power Recovery

a. Inspection Scope

The team selected complex operator actions to recover AC power to the safety related buses via the alternate AC (AAC) power source (gas turbines 1 and 3). This action should be completed within one hour and potential consequence of failure of this action is core damage after the station batteries deplete. The incorporation of this action into site procedures, classroom training, and simulator training was reviewed. The team also walked down startup and transfer of the AAC power source to a safety related 480Vac bus with operators to verify that Entergy could restore AC power within the station blackout coping duration. Finally, the team observed an operating crew respond to and implement procedures for a station blackout scenario in the simulator.

b. Findings

No findings of significance were identified.

.2.2.2 Align Backup Nitrogen Supply to Atmospheric Dump Valves

a. Inspection Scope

The team selected the operator action to locally align a backup nitrogen supply to the steam generator atmospheric dump valves (ADV). The time available to align backup nitrogen to the ADVs is critical and is based on the time at which reactor coolant system cooldown is needed to mitigate a loss of reactor coolant pump seal cooling during a station blackout event. About 60 minutes is available to diagnosis the problem and complete the local operation. The team reviewed the incorporation of this action into emergency and abnormal operating procedures, job performance measures, and training. The team observed an operator locate the local nitrogen supply valves and controls and walk through the actions to locally operate the ADVs.

b. Findings

No findings of significance were identified.

.2.2.3 Manually Restart a Component Cooling Water Pump

a. Inspection Scope

The team selected the operator action to manually restart the component cooling water (CCW) pumps given an inadvertent trip signal. This action is time critical to prevent a reactor coolant pump seal loss-of-coolant accident from occurring. The time available to restart a CCW pump is about 13 minutes. The team verified that central control room annunciator response procedures provided instructions to reset CCW pump breaker lockout relays. The team also reviewed licensed operator training plans to verify that operators would understand the lockout device associated with CCW pump breaker operation. The team observed an operating crew respond to component cooling water malfunctions in the simulator. Although the simulator scenario was not identical to this operator action, it was observed to verify that operators were alert to loss of CCW concerns and were proficient with switch manipulations to restore CCW pumps.

b. Findings

No findings of significance were identified.

.2.2.4 Align Alternate Safe Shutdown Equipment Following Switchgear Room Unavailability

a. Inspection Scope

The team selected the operator action to manually align the alternate safe shutdown systems (ASSS) in the event the control room was rendered unavailable or normal controls and indications inoperable. Such conditions could exist for fire or flooding events. Some ASSS alignments are time critical, about one hour, and the IP 2 Nuclear Power Plant Probabilistic Safety Assessment, Appendix H, Human Reliability Analysis Notebook, Revision 0, considered the operator actions as extremely high stress actions. The team observed an operator walk through the actions to align the No. 21 auxiliary feedwater pump, the No. 23 charging pump, and the No. 21 safety injection pump for alternate safe shutdown. The team verified that Entergy staged all necessary equipment and tools in an appropriate location to expeditiously align the ASSSs. The incorporation of this action into site procedures, classroom training, and job performance measures were also reviewed.

b. Findings

No findings of significance were identified.

.2.2.5 Manually Control Turbine Driven Auxiliary Feedwater Pump Following Battery Depletion

a. Inspection Scope

The team selected the operator action to manually control the turbine-driven auxiliary feedwater pump (TDAFWP) following battery depletion. The potential consequence of failure of this action is core damage after steam generators overfill and damage the TDAFWP due to moisture carryover. This operator action involved controlling several steam and feedwater valves associated with the TDAFWP. The IP 2 Nuclear Power Plant Probabilistic Safety Assessment, Appendix H, Human Reliability Analysis Notebook, Revision 0, considered the operator actions as extremely high stress with moderate complexity. The team observed an operator walk through the actions to locally control steam generator levels as well as locally operating all steam control valves to the TDAFWP. The team verified that Entergy staged all necessary tools in an appropriate location to expeditiously operate the TDAFWP. The incorporation of this action into site procedures, classroom training, and job performance measures were also reviewed.

b. Findings

No findings of significance were identified.

.3 Review of Industry Operating Experience (OE) and Generic Issues (6 Samples)

a. Inspection Scope

The team reviewed selected OE issues for applicability at Indian Point Unit 2. The team performed a detailed review of the OE issues listed below to verify that Entergy had appropriately assessed potential applicability to site equipment and initiated corrective actions when necessary.

.3.1 NRC Information Notice (IN) 1991-51, Inadequate Fuse Control Programs

The team reviewed Entergy's disposition of IN 1991-51. This Information Notice emphasized the importance of programs to control activities related to fuses. The team interviewed the IP 2 Fuse Control Program Coordinator to discuss the development of the program, implementation of the program, coordination between operations and engineering, current issues, and historical issues. The team reviewed the documents which implement the program to verify that fuses are controlled in accordance with the directives and procedures. A sample of condition reports was also reviewed to verify that fuse related problems were identified and handled appropriately.

.3.2 NRC IN 2005-023, Vibration-Induced Degradation of Butterfly Valves

The team reviewed Entergy's evaluation of IN 2005-03 to assess the thoroughness and adequacy of the subject evaluation. IN 2005-03 focused on separation of butterfly valve internal components due to the vibration-induced loss of taper pins used to connect them. Entergy's evaluation included conducting a search of the corrective action database to identify whether there were condition reports involving related valve failures, and reviewing valve preventive maintenance procedures to evaluate the measures employed at IP 2 to secure the valve disc-to-stem taper pins. The results of Entergy's evaluation indicated that the subject butterfly valves were not susceptible to vibration-induced failure as described in the Information Notice.

.3.3 NRC IN 2006-03, Motor Starter Failures Due to Mechanical-Interlock Binding

The team reviewed Entergy's disposition of IN 2006-03, which addressed mechanical-interlock binding due to misalignment that resulted from a mounting hole offset in certain motor starters. Although the subject starters were not used at IP 2, the motor starters in use have also exhibited similar problems due to mechanical-interlock binding that was determined by Entergy to be a result of age-related lubrication degradation. The team interviewed the system engineer responsible for implementing the corrective action to replace the installed starters' mechanical-interlock mechanisms with a type of an improved design. The team confirmed that all potentially vulnerable starters have been updated with the improved design mechanism for the mechanical-interlock.

.3.4 NRC IN 2006-29, Potential Common Cause Failure of Motor-Operated Valves as a Result of Stem Nut Wear

The team conducted interviews with the MOV engineers and reviewed documents in order to determine whether Entergy experienced stem nut wear issues and if appropriate action had been taken as a result of the IN. The team determined that Entergy lubricates their valve stems every two years, has not had any stem nut failures, and sends their stem nut to the valve manufacturer for machining. As a result of the IN, Entergy revised their diagnostic test procedure to evaluate stem nut wear.

.3.5 NRC IN 2006-15, Vibration Induced Degradation and Failure of Safety-Related Valves

The team reviewed Entergy's response and actions that addressed the applicability of the valve vibration issues identified in NRC IN 2006-15. The team verified that Entergy performed a review of plant equipment databases and documents, which confirmed that the station was not vulnerable to the type of valve degradation described in the Information Notice.

.3.6 NRC IN 2005-11: Internal Flooding/Spray-Down of Safety-Related Equipment Due to Unsealed Equipment Hatch Floor Plugs and/or Blocked Floor Drains

The team reviewed Entergy's disposition of IN 2005-11, which illustrated the potential for degradation of multiple trains of emergency core cooling systems as a consequence of potential flooding in safety-related areas. The team verified that Entergy entered IN 2005-11 into its corrective action program for review and considered all actions listed within the IN. Entergy actions completed included verifying that current plant configuration of flood protection features was consistent with the design basis and UFSAR descriptions, and establishing drain system periodic maintenance.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Problem Identification and Resolution

a. Inspection Scope

The team reviewed a sample of problems that were identified by Entergy and entered into the corrective action program. The team reviewed these issues to verify an appropriate threshold for identifying issues, and to evaluate the effectiveness of corrective actions related to design or qualification 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 condition reports that were sampled and reviewed by the team are listed in the attachment to this report.

b. Findings

No findings of significance were identified in addition to the corrective action deficiencies identified separately in this inspection report.

4AO6 Meetings, Including Exit

On February 15, 2007, the team presented the inspection results to Mr. J. Ventosa, Director, Engineering, and Mr. J. Comiotes, Director, Nuclear Safety Assurance, and other members of Entergy staff. The team verified that no proprietary information is documented in the report.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

R. Altadonna, Program and Components Engineer
E. Anderson, Design Engineer (Electrical)
V. Andreozzi, System Engineering Supervisor
J. Bencivenga, Design Engineer (Mechanical)
J. Bubniak, Design Engineer (Mechanical)
D. Carleton, Maintenance Supervisor
T. Chan, System Engineer
G. Dahl, Licensing Engineer
J. Etzweiler, Operations Coordinator
A. Galati, Design Engineer (Mechanical)
D. Gaynor, Senior Lead Engineer
J. Herrera, System Engineer
M. Imai, System Engineer
J. Kayani, Heat Exchanger Component Engineer
M. Kempski, System Engineer
E. Kenney, MOV Program Engineer
A. King, Design Engineer
C. Laverde, MOV Program Engineer
R. Lee, Design Engineer (Mechanical)
T. Moran, Check Valves Program Engineer
T. Orlando, Design Engineering Manager
J. Pineda, System Engineer
J. Raffaele, Design Engineering Supervisor
V. Rizzo, AOV Program Engineer
H. Robinson, Design Engineer (Electrical)
F. Weinert, Design Engineer (Electrical)
J. Whitney, System Engineer
S. Wilkie, Fire Protection Engineer

NRC Personnel

M. Cox, Senior Resident Inspector
B. Wittick, Resident Inspector
W. Schmidt, Senior Risk Analyst

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000247/2007007-03	URI	Use of Motor Control Center Methodology for Periodic Verification of the Design Basis Capability of Safety-Related MOVs (Section 1R21.2.1.2.b2)

Closed

05000247/2006005-03	URI	Reliability / Unavailability of the Gas Turbine System and Impact on Functionality (Section 1R21.2.1.10.b1)

Opened and Closed

05000247/2007007-01	NCV	Inadequate Design Control Associated with Vortexing and Net Positive Suction Head Calculations (Section 1R21.2.1.1b)
05000247/2007007-02	NCV	Inadequate Differential Pressure Value Used for MOV 746 and MOV 747 to Ensure Valve Capability (Section 1R21.2.1.2.b1)
05000247/2007007-04	NCV	Inadequate Design Control for Environmental Effects to Ensure the Availability of the Turbine Driven Auxiliary Feedwater Pump Operation (Section 1R21.2.1.7b)
05000247/2007007-05	NCV	Failure to Adequately Monitor Gas Turbine System Performance as Required by the Maintenance Rule (Section 1R21.2.1.10.b1)
05000247/2007007-06	FIN	Failure to Correct Degraded Gas Turbine 1 Reliability (Section 1R21.2.1.10.b2)
05000247/2007007-07	NCV	Inadequate Station Battery Capacity Testing for Degradation Monitoring (Section 1R21.2.1.13.b1)
05000247/2007007-08	NCV	Ineffective Corrective Action for High Inter-Tier Battery Resistances (Section 1R21.2.1.13.b2)
05000247/2007007-09	NCV	Untimely Corrective Actions for Decrease in Battery Margin (Section 1R21.2.1.13.b3)

LIST OF DOCUMENTS REVIEWED

Calculations

18.03.F02.007, Air Operated Gate/Globe Valve Component Calculations, Rev. 1
CN-SEE-03-5, IP 2 RHR Cooldown Analysis for the 5% Power Uprate Program, Rev. 0
DOE-2001-2958-FFX, Determination of Equivalency for GT1 Starting Diesel Battery, Rev. 0
EGE-00022-01, IP 2 DB-75 Breaker Overload Capability (Degraded Voltage), Rev. 1
EGP-00011-03, DC Load Study Battery 21, Rev. 3
EGP-00013-03, IP 2 DC Load Study Battery 23 Calculation, Rev. 3
FCX-00035, Operability Analysis of Stairwell No. 4 Fire Protection Piping, Rev. 0
FCX-00086-00, AFW Pump Room Temperature Rise, Rev. 0
FEX-00002-00, Develop Loading Margin for Batteries Future Loading, Rev. 1
FEX-00019-02, 118Vac Instrument Bus Loading and Voltage Drop for Buses 21&21A, Rev. 2
FEX-000205-00, 125Vdc Battery 23 Sizing and Voltage Drop, Draft
FEX-00021-02, 118Vac Instrument Bus Loading and Voltage Drop for Buses 23&23A, Rev. 2
FEX-00039-02, EDG Loading Study, Rev. 2
FEX-00044-02, 125Vdc Battery 21 Minimum Voltage Analysis, Rev. 2
FEX-00047-02, 125Vdc Battery 24 Minimum Voltage Analysis, Rev. 2
FEX-00048-02, Minimum Voltage Analysis for 125Vdc Power Panels, Rev. 2
FEX-00049-01, Battery 21 Sizing Calculation, Rev. 2
FEX-00050-02, 125Vdc Battery Sizing Calculation, Rev. 2
FEX-00051-01, 125Vdc Battery 23 Sizing Calculation, Rev. 1
FEX-00053-01, 125Vdc Battery 21 Voltage Profile, Rev. 1
FEX-00058-00, Verification of the Charge Time Adequacy of Battery Charger 21, Rev. 0
FEX-00062-01, Minimum Operating Electrolyte Temperature for 125Vdc Batteries, Rev. 1
FEX-00071-00, Indian Point Analysis of Loading on 125Vdc Batteries During SBO, Rev. 0
FEX-00143-01, IP 2 Load Flow Analysis of the Electrical Distribution System, Rev. 1
FEX-00180-00, MCC Control Circuit Voltage Evaluation, Rev. 0
FFX-0023, Fire Piping Outside the 480Vac Switchgear Room for Seismic Loading, Rev. 0
FIX-00004, Motor-Driven AFW Pump Flow Loop Accuracy, Rev. 1
FIX-00024-02, CST - Level Setpoints, Channel Accuracies, and Volumes, Rev. 2
FIX-00027-01, EDG Coolers - SW Flow Alarm and Controller Accuracies, Rev. 1
FIX-00029-00, EDG Fuel Oil Day Tank Level Control/Alarm Instrument Uncertainty, Rev. 0
FIX-00056, Overpressure Protection System Instrumentation Loop, Rev. 3
FIX-00068-00, SI Pump Miniflow Line Flow Indicator FI-950 Accuracy, Rev.0
FIX-00069, AFW Pump Instrument Accuracy and ASME Section XI Testing, Rev. 3
FIX-00073-00, Pressurizer Cold Calibrated Level Indication Uncertainties, Rev. 0
FIX-00138, IP 2 ITS Allowable Value - 480Vac Bus UV & Degraded Voltage, Rev. 0
FMX-00050-01, High Head SI Pump Available NPSH, Rev. 1
FMX-00085-00, RWST Minimum Submergence Level, Rev.0
FMX-00086-01, CST Submergence at Varied Flow Rates, Rev. 1
FMX-00128-00, EDG - Jacket Water Cooler/Lube Oil Cooler Bundle Replacement, Rev. 0
FMX-00227-01, Pipe Flow Calculation of SW System, Rev. 1
FMX-00236-01, SW Low Pressure Alarm Setpoint, Rev. 1
FMX-00270-00, IP 2 OPS Thermal Hydraulic Analysis, Rev. 0
FMX-00275-01, Pipe Flow Analysis for AFW System, Rev. 1

FMX-00287, Verification of AFW Pump Recirculation Flow Test Acceptance Criteria, Rev. 1
FMX-00295-00, Tube Plugging Limits for EDG Lube Oil and Jacked Water Coolers, Rev. 0
FMX-00324, RWST Vent Verification, Rev. 0
FMX-00353-00, NPSH Available for Service Water Pumps, Rev. 0
FPX-00281-00, Nitrogen Capacity Requirements of PORV Accumulators, Rev. 0
GSX-00036-00, IP 2 EOP Setpoint and CST/RWST Level EOP Setpoint Analysis, Rev. 0
IP-CALC-04-01062, IP 2 SW System Heat Load, Rev. 0
IP-CALC-04-01589, Load Flow and Short Circuit Analysis - Electrical Distribution, Rev. 0
IP-CALC-06-00158, Analysis of Motor Torque for MCC Testing of MOV 784, Rev. 0
IP-CALC-06-00329, Replacement of EDG Air Start Motors, Rev. 0
IP-CALC-06-00372, IP 2 EDG Fuel Oil Storage Requirements, Rev. 0
JO 9321-01, EDG Building Ventilating, 1/23/68
JOG-TD-01, Spring Relaxation for Air Operators, Rev. 1
MEX-00131, Evaluation of GL 89-10 MOVs for Pressure Locking and Thermal Binding, Rev. 0
MMM-00011-00, EDG Fuel Oil Transfer Pump Submergence, Rev. 0
MMS-00065, Analysis of Thrust and Torque Limits for MOV 536, Rev. 9
MMS-00088, Analysis of Thrust and Torque Limits for MOV 747, Rev. 12
MMS-00116-06, Analysis of Thrust and Torque Limits for MOV 885A, Rev. 6
NSL-EDG-900430A, EDG Fuel Oil Minimum Storage Requirements, 10/12/90
PE-SW-910830A, SW Pump Submergence and NPSH, Rev. 0
PGI-00051, MOV 885A&B Differential Pressure Calculation, Rev. 1
PGI-00059, 746&747 Differential Pressure Calculation, Rev. 2
PGI-00076-00, Charging Pump Operation (Net Positive Suction Head - Available), Rev. 0
PGI-00087-00, EDG Lube Oil Cooler Sizing, Rev. 1
PGI-00173-00, Ventilation of Primary Auxiliary Building, Rev. 0
PGI-00218-00, External Recirculation and RWST Leakage Allowances, Rev. 0
PGI-00287, Evaluation of MOV Load Sensitive Behavior, Rev. 1
PGI-00288, Evaluation of MOV Static and Dynamic Coefficient of Friction, Rev. 1
PGI-00289, Evaluation of MOV Stem Lubrication Degradation, Rev. 1
PGI-00331, MOV Evaluation for 885A and 885B, Rev. 2
PGI-00332, MOV Evaluation for 746 and 747, Rev. 2
PGI-00349, Evaluation of MOV Test Equipment Accuracy, Rev. 1
PGI-00350, Valve Factor Basis Evaluation, Rev. 2
PGI-00351, Stem Packing Load Basis Document, Rev. 0
PGI-00472, 480Vac MCC Bus Degraded Voltage Altran Calculation 01004-C-001, Rev. 2
PGI-00473, Motor Operated Valve Terminal Voltage Altran Calculation 99621-C-002, Rev. 3
PGI-00475, GL-89-10 MOV Protection - TOR Settings, Rev. 2
PGI-00497, AFW AOV Functional and Differential Pressure Calculation, Rev. 1
SAE/FSE-C-IPP-0138, IP 2 AFW System Model and System Performance, Rev. 0
SEE-06-09, IP 2 SI Pump Discharge Valve Sump Debris Evaluation, Rev. 0
SGX-00007-03, 125Vdc Protective Device Coordination Study, Rev. 3
SGX-00013-05, Setpoint Change for UV Relays on 480 Volt Buses, Rev. 5
SGX-00047-00, Replacement of MOV - Thermal Overload Sizing Calculation, Rev. 0
SGX-00048-01, IP 2 480Vac Switchgear Coordination Calculation, Rev. 1

Completed Surveillance Test Procedures

0-VLV-404-AOV, Use of Air Operated Valve Diagnostics (6/7/06)
2-IC-PC-I-L-1206S, EDG Fuel Oil Storage Tank No. 23 Level (1/23/07)
2-IC-PC-I-L-1209S, EDG Fuel Oil Day Tank No. 23 Level (1/23/07)
2-IC-PC-N-SAT, Station Auxiliary Transformer Instruments (5/2/06)
2-PC-R58, 480Vac UV Relay Calibration (2/17/05)
2-PT-Q001A, 21 Station Battery Surveillance and Charging (8/21/06, 11/13/06)
2-PT-Q013, Inservice Valve Tests for MOV 885A and MOV 885B (2/10/06)
2-PT-Q024C, 23 EDG Fuel Oil Transfer Pump (5/21/06, 8/09/06, 11/03/06)
2-PT-Q026A, 21 Service Water Pump (6/4/06, 8/29/06)
2-PT-Q026B, 22 Service Water Pump (12/17/06)
2-PT-Q026C, 23 Service Water Pump (12/17/06)
2-PT-Q026D, 24 Service Water Pump (1/9/07)
2-PT-Q026E, 25 Service Water Pump (11/17/06)
2-PT-Q026F, 26 Service Water Pump (11/17/06)
2-PT-Q027A, 21 Auxiliary Feed Pump (5/25/06, 8/16/06, 11/06/06)
2-PT-Q029A, 21 Safety Injection Pump (6/2/06, 6/21/06, 9/14/06, 12/8/06)
2-PT-Q33C, 23 Charging Pump (5/31/06, 8/02/06, 10/29/06, 11/07/06)
2-PT-R007A, Motor Driven AFW Pumps Full Flow (10/24/02, 10/19/04, 4/17/06)
2-PT-R014, Automatic SI System Electrical Load and Blackout Test (4/22/06)
2-PT-R022A, Steam Driven AFW Pump Full Flow (11/23/02, 11/19/04, 4/17/06)
2-PT-R029, Safety Injection Check Valves, (11/3/02, 10/31/04, 5/7/06)
2-PT-R082, RCS OPS Nitrogen System Check (11/5/02, 11/8/04, 5/5/06)
2-PT-R093, Essential SW Header Flow Balance (11/16/04)
2-PT-V024, IST Tests for MOV 746 and MOV 747 (10/30/02, 10/24/04, 5/15/06, 10/13/06)
2-PT-V067, Essential Service Water Header Flow Balance (11/16/04)
2-PT-W010, Weekly Battery Surveillance Requirement (1/10/07, 1/3/07)
2-PT-W020, Weekly Inverter Verification (1/6/07)
2-VLV-017-MOV, Acquiring/Analyzing MCC Data (MOVATS) for MOV 747 (9/3/04)
GT-1 Reliability Checklists from SOP 31.1.1 Attachment 1 (numerous from 2004-2007)
MOV-B-013-A, MOV Static Test Evaluation for MOV 885A (3/20/98)
PC-Q 2, RWST Level (1/24/05, 12/27/05, 12/28/06)
PC-R53, AFW Pump Room EQ Temperature Switches (3/30/95, 6/16/97, 9/16/02, 3/5/05)
PM No. 1784, EDG Fuel Oil Storage/Day Tank No. 23 Levels (1/27/98, 11/9/02, 11/10/02)
PT-2Y11A, Gas Turbine 1 Blackstart Timing (6/12/03, 4/19/05)
PT-2Y11C, Gas Turbine 3 Blackstart Timing (9/9/02, 4/25/04, 1/2/06)
PT-A35A, 21 Station Battery Inter-cell Resistance Checks (3/4/04, 2/2/05, 1/4/06, 12/4/06)
PT-M22, Station Battery Monthly Surveillance (11/10/07, 12/14/06)
PT-M38A, Gas Turbine No. 1 (12/19/06)
PT-M63A, Gas Turbine 1 Batteries (1/8/07)
PT-R76A, Station Battery 21 Load (11/5/02, 11/3/04, 4/25/06)
PT-V42, Gas Turbine Blackstart Timing (11/2/93, 11/4/93, 11/18/93)
SE-SQ-12.314, MOV Static Test Evaluation for MOV 747 (5/10/00)

Condition Reports

2000-02947	2005-04671	2006-06007	2007-00309	2007-00656*
2000-06049	2005-04875	2006-06227	2007-00356	2007-00659*
2000-08952	2005-04908	2006-06249	2007-00390*	2007-00662*
2000-09882	2005-05324	2006-06636	2007-00404*	2007-00679*
2000-10850	2006-00003	2006-06712	2007-00408*	2007-00681*
2001-00363	2006-00023	2006-06732	2007-00409*	2007-00684*
2001-00970	2006-00043	2006-06850	2007-00419*	2007-00695*
2001-03128	2006-00200	2006-06939	2007-00420*	2007-00702*
2002-02994	2006-00314	2006-07238	2007-00429*	2007-00712*
2003-06088	2006-00379	2007-00008	2007-00432*	2007-00715*
2004-01056	2006-00631	2007-00034	2007-00437*	2007-00716*
2004-02805	2006-00640	2007-00105	2007-00439*	2007-00720*
2004-04277	2006-01299	2007-00124*	2007-00440*	2007-00737*
2004-05820	2006-01689	2007-00125*	2007-00448*	2007-00749*
2004-06711	2006-02020	2007-00133*	2007-00452*	2007-00777*
2005-00453	2006-02256	2007-00163*	2007-00463*	2007-00780*
2005-01450	2006-03094	2007-00193*	2007-00487*	2007-00803*
2005-01944	2006-03531	2007-00225	2007-00517*	2007-00826*
2005-02098	2006-04720	2007-00236	2007-00525	2007-00836*
2005-02887	2006-04739	2007-00259	2007-00539*	2007-00839*
2005-03948	2006-05787	2007-00274	2007-00641*	2007-00842*
2005-04137	2006-05793	2007-00308	2007-00651	2007-00847*

LO-OEN-2005-00193
LO-OEN-2006-00043

* Condition Report was written as a result of inspection effort.

Work Orders

04-14609	05-22867	03-30217	02-63040	04-31533
04-33609	05-22869	02-60369	03-21135	05-18105
01-22068	04-28954	02-58483	03-21136	05-20705
01-21964	06-23642	02-25609	03-21267	05-25438
03-14382	99-07492	99-07501	03-24752	06-23159
04-24880	99-07493	00-18327	04-15181	07-12315
04-35854	04-31008	02-63061	04-20210	

Drawings

144D882, (480Vac SWGR 21 & 22) Field Modifications to Unit #20, Rev. 0
242688, Flow Diagram Instrument Air Containment Bldg & AFW Pump Building, Rev. 25
311907, Wiring Diagram PORV 456, Rev. 1
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2-SOP-1.4.1, Overpressure Protection System Operation, Rev. 19
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LIST OF ACRONYMS USED

AAC	Alternate AC
AC	Alternating Current
ADV	Atmospheric Dump Valve
AFW	Auxiliary Feedwater
AFWP	Auxiliary Feedwater Pump
AOT	Allowed Outage Time
ASSS	Alternate Safe Shutdown System
CCW	Component Cooling Water
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CR	Condition Report
DC	Direct Current
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
GL	[NRC] Generic Letter
gpm	Gallons per Minute
GT	Gas Turbine
IEEE	Institute of Electrical and Electronics Engineers
IMC	Inspection Manual Chapter
IN	[NRC] Information Notice
IP 2	Indian Point Unit 2
LOCA	Loss-of-Coolant Accident
LOOP	Loss-of-Offsite Power
LTOP	Low Temperature Overpressure Protection
MCC	Motor Control Center
MOV	Motor Operated Valve
MPFF	Maintenance Preventable Functional Failure
NCV	Non-Cited Violation
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
OE	Operating Experience
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Analysis

psid	Pounds per Square Inch (Differential)
psig	Pounds per Square Inch (Gauge)
RAW	Risk Achievement Worth
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RMPFF	Repeat Maintenance Preventable Functional Failure
ROP	Reactor Oversight Process
RRW	Risk Reduction Worth
RWST	Refueling Water Storage Tank
SDP	Significance Determination Process
SI	Safety Injection
SPAR	Standardized Plant Analysis Risk
SW	Service Water
TDAFWP	Turbine Driven Auxiliary Feedwater Pump
TS	Temperature Switch
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
Vac	Volts Alternating Current
Vdc	Volts Direct Current