UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
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ATLANTA, GEORGIA 30303-8931

August 17, 2009

Mr. Dennis R. Madison
Vice President
Southern Nuclear Operating Company, Inc.
Edwin I. Hatch Nuclear Plant
11028 Hatch Parkway North
Baxley, GA 31513

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT - NRC COMPONENT DESIGN BASES INSPECTION - INSPECTION REPORT 05000321/2009006 AND 05000366/2009006

Dear Mr. Madison:

On June 5, 2009, U. S. Nuclear Regulatory Commission (NRC) completed an inspection at your Edwin I. Hatch Nuclear Plant, Units 1 and 2. The enclosed inspection report documents the inspection results, which were discussed on June 5, and July 20, 2009, with Mr. Sonny Bargeron and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The team reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents four NRC-identified findings of very low safety significance which were determined to be violations of NRC requirements. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy because of their very low safety significance and because they were entered into your corrective action program. If you contest these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC resident inspector at the Edwin I. Hatch Nuclear Plant. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region II, and the NRC Resident Inspector at the Hatch Nuclear Plant. The information you provide will be considered in accordance with the Inspection Manual Chapter 0305.

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 (PARS) component of the 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/

Binoy B. Desai, Chief
Engineering Branch 1
Division of Reactor Safety

Docket Nos.: 50-321, 50-366
License Nos.: DPR-57 and NPF-5

Enclosure: Inspection Report 05000321/2009006, 05000366/2009006
w/Attachment: Supplemental Information

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Letter to Dennis R. Madison from Binoy B. Desai dated August 17, 2009.

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT - NRC COMPONENT DESIGN BASES INSPECTION - INSPECTION REPORT 05000321/2009006 AND 05000366/2009006

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-321, 50-366
License Nos.:	DPR-57 and NPF-5
Report Nos.:	05000321/2009006, 05000366/2009006
Licensee:	Southern Nuclear Operating Company, Inc.
Facility:	Edwin I. Hatch Nuclear Plant
Location:	Baxley, Georgia 31513
Dates:	May 4 – July 20, 2009
Inspectors:	D. Jones, Senior Reactor Inspector (Lead) R. Lewis, Senior Reactor Inspector J. Hamman, Reactor Inspector R. Williams, Reactor Inspector J. Eargle, Reactor Inspector A. Sengupta, Reactor Inspector (Trainee) C. Smith, Construction Inspector (Trainee) G. Skinner, Contractor C. Baron, Contractor
Approved by:	Binoy B. Desai, Chief Engineering Branch 1 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000321/2009006, 05000366/2009006; 5/4/2009 – 6/5/2009; Edwin I. Hatch Nuclear Plant, Units 1 and 2; Component Design Bases Inspection.

This inspection was conducted by a team of five NRC inspectors from the Region II office, and two NRC contract inspectors. Four findings of very low significance (Green) were identified during this inspection and were classified as 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," (ROP) Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The team identified a non-cited violation of 10 CFR 50, Appendix B, Criterion XI, Test Control, for failure to correctly establish acceptance criteria for the Standby Diesel Service Water (SDSW) System. The licensee performed a past operability determination and initiated Condition Report (CR) 2009105651 to revise the acceptance criteria.

  The licensee's failure to correctly establish acceptance criterion for the SDSW pump under the most limiting conditions was a performance deficiency. The finding is greater than minor because it adversely affected the Equipment Performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The finding is of very low safety significance (Green) using the SDP because it did not represent a loss of system or safety function. A cross-cutting aspect was not identified because the finding does not represent current performance. [Section 1R21.2.8]

- Green. The team identified a non-cited violation of 10 CFR 50.65(a)(1) for the licensee's failure to monitor the main steam line and feedwater line pipe whip restraints for Units 1 and 2. The licensee initiated CRs 2009105147 and 200910622 and plans to complete inspections of the whip restraints during the upcoming Units 1 and 2 outages.

  The licensee's failure to periodically inspect the condition of the safety-related pipe whip restraints was a performance deficiency. The finding is more than minor because it is associated with Equipment Performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The team determined that the finding is of very low safety significance (Green) using the SDP because the finding did not represent an actual loss of safety function. The finding directly involved the cross-cutting aspect of implementing a corrective action program with a low threshold for identifying issues under the Corrective Action Program component of the Problem Identification and Resolution area [P.1(a)]. [Section 1R21.2.9]

Cornerstone: Barrier Integrity

- Green. The team identified a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, for the licensee's failure to promptly correct deficiencies in containment penetration seals. The licensee initiated CR 2009105747 to evaluate corrective actions for the seals.

  The team determined that the failure to take corrective actions for deficiencies in containment penetration seals was a performance deficiency. The finding is greater than minor because it is associated with the Structures, Systems and Components (SSC) and Barrier Performance attribute of maintaining functionality of containment and affected the cornerstone objective of providing reasonable assurance that containment protects the public from radionuclide releases caused by accidents or events. The finding is of very low safety significance (Green) because the finding did not represent an actual open pathway in the physical integrity of reactor containment. The finding directly involved the cross-cutting aspect of thoroughness of evaluation within the Corrective Action Program component of the Problem Identification and Resolution area [P.1(c)]. [Section 1R21.2.1]

- Green. The team identified a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control, for failure to correctly establish containment isolation valve leakage criteria for Unit 2 feedwater check valves. The licensee initiated CR 2009104567 and revised the associated calculation during the inspection.

  The team determined that the failure to correctly establish leakage acceptance criteria for the feedwater check valves was a performance deficiency. The finding is greater than minor because it is associated with the SSC and Barrier performance attribute of maintaining functionality of containment and affected the cornerstone objective of providing reasonable assurance that containment protects the public from radionuclide releases caused by accidents or events. The finding is of very low safety significance (Green) because the finding did not represent an actual open pathway in the physical integrity of reactor containment. The finding directly involved the cross-cutting aspect of complete, accurate and up-to-date design documentation within the Resources component of the Human Performance area[H.2(c)]. [Section 1R21.2.2]

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R21 Component Design Bases Inspection (71111.21)

.1 Inspection Sample Selection Process

The team selected risk significant components and operator actions for review using information contained in the licensee's Probabilistic Risk Assessment (PRA). In general, this included components and operator actions that had a risk achievement worth factor greater than 1.3 or Birnbaum value greater than 1 $X10^{-6}$. The components selected were primarily located within systems that mitigate high energy line breaks, and internal flooding events. The sample included sixteen components, five operator actions, and three operating experience items.

The team performed a margin assessment and detailed review of the selected risk-significant components to verify that the design bases had been correctly implemented and maintained. This design margin assessment considered original design issues, margin reductions due to modifications, or margin reductions identified as a result of material condition issues. Equipment reliability issues were also considered in the selection of components for detailed review. These reliability issues included items related to failed performance test results, significant corrective action, repeated maintenance, maintenance rule (a)1 status, RIS 05-020 (formerly GL 91-18) conditions, NRC resident inspector input of problem equipment, System Health Reports, industry operating experience and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense in depth margins. An overall summary of the reviews performed and the specific inspection findings identified is included in the following sections of the report.

.2 Results of Detailed Reviews

.2.1 Unit 2 Mechanical and Electrical Primary Containment Penetrations

a. Inspection Scope

The team reviewed the safety analysis report, technical specifications, technical specification bases, and drawings to identify the design bases requirements of the Unit 2 primary containment penetrations. The team examined system health reports, records of surveillance testing, and applicable corrective actions to verify that potential degradation was being monitored and prevented or corrected. The team also reviewed industry guidance and applicable plant procedures to verify that degraded conditions had been appropriately evaluated and corrected. The team performed interviews with plant personnel to discuss the history of containment penetration surveillance testing and the evaluations and corrective actions that resulted from the testing. The team also reviewed equivalent Unit 1 records to determine if the Unit 1 penetrations had similar surveillance test results.

b. Findings

Introduction: The NRC identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, for failure to promptly correct deficiencies in containment penetration seals.

Description: The team reviewed Condition Report (CR) 2009101411, initiated February 16, 2009, which documented Local Leak Rate Test (LLRT) failures that occurred during the 2009 Unit 2 outage. CR 2009101411 referenced nine CRs that addressed LLRT failures of two electrical and seven mechanical penetrations in 2009. For the nine penetrations that exceeded their LLRT administrative limits, the CRs determined that repair was not required because the collective leakage of all penetrations did not exceeded the primary containment overall leakage rate acceptance criteria (Technical Specification 5.5.12). In addition to not performing repairs, the licensee did not perform evaluations to determine the cause of the degraded conditions. The team's review of LLRT results revealed that a number of Unit 1 and 2 penetrations had repeatedly exceeded their administrative limit between 1994 and 2009 without being repaired or evaluated.

The team noted that Technical Specification 5.5.12, Primary Containment Leakage Rate Test Program, states that the program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, Performance-Based Containment Leak-Test Program, dated September, 1995 and NEI 94-01, Rev. 0, Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J. NEI 94-01, Section 10.2.1.4, Corrective Action, states that a cause determination should be performed and corrective actions identified that focus on those activities that can eliminate the identified cause of failure with appropriate steps to eliminate recurrence. Additionally, Section 10.2.1.4 states, failures include exceeding an administrative limit, not just the total failure of a penetration.

Analysis: The team determined that the failure to take corrective actions for deficiencies in containment penetration seals was a performance deficiency. The team determined that the finding is greater than minor because it is similar to example 4.a of MC 0612, Appendix E in that licensee routinely failed to perform apparent cause determinations on similar issues, and it is associated with the Structures, Systems and Components (SSC) and Barrier performance attribute of maintaining functionality of containment and affected the cornerstone objective of providing reasonable assurance that containment protects the public from radionuclide releases caused by accidents or events. The team determined that the finding is of very low safety significance (Green) using the SDP because the finding did not represent an actual open pathway in the physical integrity of reactor containment. The finding directly involved the cross-cutting aspect of thoroughness of evaluation within the Corrective Action Program component of the Problem Identification and Resolution area. Specifically, the licensee failed to evaluate the repetitive exceedance of LLRT administrative limits when the issues were entered into the corrective action program. [P.1(c)].

Enforcement: 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, states in part, that measures shall be established to assure that conditions adverse to quality, such as deficiencies are promptly identified and corrected. Contrary to the above, the licensee failed to promptly correct conditions adverse to quality when deficiencies were identified during testing. Specifically, from 1994 – 2009 when LLRT identified conditions adverse to quality, the licensee did not correct deficiencies in containment penetration seals. Because this finding is of very low safety significance and because it was entered into the licensee's corrective action program as condition report 2009105747, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000321/2009006-01 and 05000366/2009006-01, Failure to Perform Cause Determinations and Corrective Actions for Deficiencies in Containment Penetration Seals.

.2.2 Feedwater Check Valves 2B21-F010A/B

a. Inspection Scope

The team reviewed the safety analysis report, technical specifications, applicable plant calculations, and drawings to identify the design bases requirements of the Unit 2 feedwater check valves. The team examined valve modifications to determine the impact of the changes on the valves' functions. The team examined system health reports, records of surveillance testing, maintenance and modification activities, and applicable corrective actions to verify that potential degradation was being monitored and prevented or corrected. The team reviewed Licensee Event Reports (LER) and root cause evaluations that had been generated as a result of leakage test failures as well as operability evaluations that had been prepared to address the condition of the valves at the time of the inspection. The team performed a detailed review of the design calculation prepared to establish the valves' leakage limits to verify that the calculated limits were adequate to ensure that the valves would perform their function under the most limiting accident conditions. The team also compared the Unit 2 valve design and history of surveillance test results to the equivalent Unit 1 valves to determine if the Unit 1 valves had similar surveillance test results. The team performed several interviews with plant personnel to discuss the history of valve testing, maintenance, and modifications to the Unit 2 valves.

b. Findings

Introduction: The team identified a violation of 10 CFR 50, Appendix B, Criterion III, Design Control, for failure to correctly establish containment isolation valve leakage criteria for Unit 2 feedwater check valves.

Description: The function of the feedwater check valves is to maintain the piping between the valves and the reactor vessel filled with water during accident conditions to prevent the release of radionuclides to the environment. The licensee recently performed Calculation SMNH-09-004, FW Check Valves 2B21-F010A/B & 2B21-F077A/B, to determine the maximum allowable leakage criteria for Unit 2 feedwater check valves. The calculation established leakage criteria for LLRT to assure that the piping would remain filled with water for 30 days following an accident. The leak test acceptance criteria was determined in part by calculating the volume of water that would be lost due to flashing/evaporation if the piping was depressurized following an accident.

The inspection team determined that the calculation did not correctly determine the amount of flashing/evaporation from the piping. Specifically, the calculation did not use the most limiting (lowest) post accident primary containment pressure, did not consider the energy input from the steel piping, and did not address measurement uncertainty associated with the leakage test. The licensee initiated CR 2009104567 and twice revised the calculation (Versions 2 and 3) during the inspection. The revisions significantly reduced the calculated margin between the leakage limit and the as-left condition of check valve 2B21-F010A from 17% to 2%. The licensee concluded that the valve remained operable and revised their Prompt Determination of Operability to address the significant reduction in margin.

Analysis: The team determined that the failure to correctly calculate the acceptance criteria for the feedwater check valves was a performance deficiency. The team concluded that the finding is greater than minor because it is similar to example 3.j of MC 0612, Appendix E in that it resulted in a condition where there was a reasonable doubt as to the operability of a component, and it is associated with the Structures, Systems and Components (SSC) and Barrier performance attribute of maintaining functionality of containment and affected the cornerstone objective of providing reasonable assurance that containment protects the public from radionuclide releases caused by accidents or events. The team determined that the finding is of very low safety significance (Green) using the SDP because the finding did not represent an actual open pathway in the physical integrity of reactor containment. The finding directly involved the cross-cutting aspect of complete, accurate and up-to-date design documentation within the Resources component of the Human Performance area. Specifically, in March 2009, the licensee failed to ensure that personnel and procedures were adequate to determine the acceptance criteria in a calculation. [H.2(c)].

Enforcement: 10 CFR 50, Appendix B, Criterion III states, in part, that measures shall be established to assure that applicable regulatory requirements and the design bases for those structures, systems, and components are correctly translated into specifications, drawings, procedures, and instructions. Contrary the above, March 13, 2009, the license did not correctly determine the leakage limit for the feedwater check valves and did not translate the correct value into plant procedures. Because this finding is of very low safety significance and because it was entered into the licensee’s corrective action program as condition report 2009104567, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000321/2009006-02 and 05000366/2009006-02, Failure to Correctly Establish Containment Isolation Valve Leakage Criteria for the Unit 2 Feedwater Check Valves.

.2.3 Main Steam Isolation Valves (MSIV) 2B21-F022B/22C/28B/28C

a. Inspection Scope

The team reviewed the safety analysis report, technical specifications, applicable plant calculations, and drawings to identify the design bases requirements of the Unit 2 MSIVs. The team examined system health reports, records of surveillance testing, maintenance activities, and applicable corrective actions to verify that potential degradation was being monitored and prevented or corrected. The team reviewed a LER that had been generated as a result of leakage test failures and the associated root cause evaluation that addressed the failure. The team compared the Unit 2 valve design and history of surveillance test results to the equivalent Unit 1 valves to determine if the Unit 1 valves had similar surveillance test results. The team also performed several interviews with plant personnel to discuss the history of the valve testing, maintenance, and details of the corrective actions that had been completed.

b. Findings

No findings of significance were identified.

.2.4 Residual Heat Removal (RHR) Pumps 1A and 1C

a. Inspection Scope

The team reviewed the safety analysis report, technical specifications, applicable plant calculations, and drawings to identify the design bases requirements of the Unit 1 RHR pumps. The team examined system health reports, records of surveillance testing and maintenance activities, and applicable corrective actions to verify that potential degradation was being monitored and prevented or corrected. The team investigated pump vibrations that had been experienced at specific flow rates and reviewed operability evaluations that had been prepared to address the capability of the pumps to perform their design functions. The team reviewed the plants evaluation of NRC Information Notice 2007-021 to determine if it was related to the vibration issues. The team reviewed the RHR pump net positive suction head (NPSH) design calculation to verify the pumps would have adequate suction head under accident conditions. The team also performed interviews with plant personnel to discuss the history of the vibration issue, current restrictions on pump operation, and the status of the associated corrective actions.

b. Findings

No findings of significance were identified.

.2.5 Unit 1 Residual Heat Removal (RHR) and Core Spray (CS) Room Coolers

a. Inspection Scope

The team reviewed the safety analysis report, technical specifications, applicable plant calculations, and drawings to identify the design bases requirements of the Unit 1 RHR/CS room coolers. The team examined system health reports, records of surveillance testing and maintenance activities, and applicable corrective actions to verify that potential degradation was being monitored and prevented or corrected. The team investigated the basis of the cooler test acceptance criteria to verify that periodic testing would demonstrate the capability of the coolers to perform their design function under accident conditions. The team evaluated the electrical power supplies associated with each of the coolers to verify their capability to withstand a postulated single failure. The team also performed interviews with plant personnel to discuss the condition of the room coolers.

b. Findings

No findings of significance were identified

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.2.6 Non-Interruptible Essential Instrument Air Header Check Valves (Units 1 and 2)

a. Inspection Scope

The team reviewed system descriptions, safety analysis report, technical specifications, procedures, and drawings to identify the design bases requirements of the non-interruptible instrument air check valves. The team examined maintenance rule documentation to verify that the check valves were properly scoped, and monitored. The team reviewed system health reports to verify that instrument air quality was being monitored and maintained within FSAR described limits. The team reviewed the site response for IN 2008-006 to verify appropriate response and compliance with Operating Experience procedures. The team also conducted field walkdowns of the applicable portions of the instrument air system to verify, by visual observation of reasonably accessible locations, that the installed configuration and material condition were consistent with the design bases and plant drawings.

b. Findings

Introduction: The team identified an Unresolved Item (URI) regarding non-interruptible essential instrument air header check valves for Units 1 and 2. The licensee had not performed periodic maintenance or testing that demonstrated the capability of the check valves to prevent back-flow during a loss of instrument air event.

Description: During a loss of instrument air event, the back-up nitrogen system automatically pressurizes the non-interruptible essential air system with nitrogen. The non-interruptible essential air header is designed with check valves that function as a boundary between instrument air and non-interruptible essential air. The boundary check valves prevent the loss of back-up nitrogen through postulated breaks in the instrument air system. The team noted that emergency operating procedures utilize the back-up nitrogen system for operation of the hardened containment vent, which is a dominant contributor to the plant's overall core damage frequency risk profile, during loss of instrument air events.

The licensee scoped the function of the non-interruptible essential air system into the maintenance rule program as documented in the Performance Criteria dated June 19, 1998. Since initial plant start-up of Units 1 and 2, the licensee had not performed periodic maintenance or testing that demonstrated the capability of the check valves to prevent back-flow during a loss of instrument air event. The team determined that the lack of periodic maintenance or testing resulted in a lack of reasonable assurance that the valves could perform their design function if called upon. The licensee initiated CR 2009105109 for this issue and provided interim guidance to operators for responding to a loss of instrument air event.

Summary: This issue is unresolved pending further inspection and interface with the licensee to determine the extent of condition and impact from the lack of periodic maintenance or testing of the non-interruptible essential instrument air header check valves. (URI 05000321/2009006-03 and 05000366/2009006-03, Non-Interruptible Essential Instrument Air Header Check Valves for Units 1 and 2)

.2.7 Reactor Building Equipment Drain Sump System

a. Inspection Scope

The team reviewed system descriptions, safety analysis report, technical specifications, procedures, and drawings to identify the design bases requirements of the reactor building equipment drain sump system for diagonal and torus sumps. The team examined maintenance rule documentation to verify that the system was properly scoped, and monitored. The team examined equipment history documentation to verify that the design bases had been maintained. The team examined records of surveillance testing, maintenance and modification activities, and applicable corrective actions to verify that potential degradation was being monitored and prevented or corrected. The team also conducted field walkdowns of the applicable portions of the reactor building equipment drain sump system and torus room penetration seals to verify, by visual observation of reasonably accessible locations, that the installed configuration and material condition were consistent with the design bases and plant drawings.

b. Findings

Introduction: The team identified an Unresolved Item (URI) regarding the licensee's failure to scope and monitor the Reactor Building Equipment Drain Sump System for Units 1 and 2 in the maintenance rule program.

Description: The torus room and the reactor building diagonal rooms are equipped with instrumented floor drain sumps. The diagonal rooms house the High Pressure Core Injection (HPCI), Reactor Core Injection Cooling (RCIC), Control Rod Drive, Core Spray, and RHR pumps. The instrumented sumps are isolatable from each other by means of air operated valves (AOV). Automatic closure of the normally open AOVs isolates the reactor building diagonal rooms, which prevents the spread of water from room to room. The AOVs are automatically closed when high water levels are detected by sump level switches. FSAR, Section 9.3.3, states in part that a single failure of a level switch will not prevent sump isolation from occurring.

The team determined that the licensee failed to scope and monitor the Reactor Building Equipment Drain Sump System in their maintenance rule program since 1996. The team's preliminary review of corrective action documents, surveillance records, and work orders revealed a lack of functional testing on Unit 1, repetitive failures of AOVs during weekly surveillance testing on both Units 1 and 2, and inadequate corrective actions for repetitive AOV failures. Additionally, the team's questioning revealed that the design of the level switches did not meet the single failure criteria as stated in FSAR Section 9.3.3. The team determined that the licensee's flood analysis did not account for a single failure of the level switches; therefore, the flood analysis did not evaluate the effects of flooding in the diagonal rooms of Units 1 and 2.

As a result of the team's observations, the licensee completed Engineering Response, RER C091204801, Flooding of Torus Room and Diagonals, during the inspection. RER C091204801, determined that a main feedwater line break with a postulated single failure of: Unit 1 level switch (1T45-N007) would result in the loss of RCIC system, and Unit 2 level switches (2T45-N006 or 2T45-N007), would result in the loss of the HPCI system or RCIC systems. The licensee initiated CRs (2009105744, 2009105110, 2009105111, 2009105615, and 2009105727) and administratively closed the AOVs as an interim compensatory measure.

Summary: This issue is unresolved pending further inspection and interface with the licensee to determine the extent of condition and impact from the failure to scope and monitor the Reactor Building Equipment Drain Sump System in the licensee's maintenance rule program, and the single failure design deficiency for the level switches. (URI 05000321/2009006-04 and 05000366/2009006-04, Reactor Building Equipment Drain Sump System for Units 1 and 2)

.2.8 2P41-C002 Standby Diesel Service Water (SDSW) Pump Inspection Scope

a. Inspection Scope

The team reviewed system descriptions, safety analysis report, technical specifications, and applicable plant calculations and evaluations, procedures, and drawings to identify the design bases requirements of the SDSW pump. The team examined equipment history documentation to verify that the design bases had been maintained. The team examined records of surveillance testing, maintenance and modification activities, system health reports, maintenance rule documentation, and applicable corrective actions to verify that potential degradation was being monitored and prevented or corrected. The team reviewed the availability of water from the suction source under routine service as well as the extremes of high and low water conditions. These reviews included: available water levels; water temperatures; the provision of adequate pump NPSH; submergence protection; and adequate minimum flow protection to verify that the pump was capable of performing its function under design bases conditions. The reviews also included verification of related aspects of flow models and calculations which demonstrated the capability of the pump to provide system flows and developed heads in accordance with design bases requirements in servicing the emergency diesel generator cooling loads. The team reviewed the licensee's establishment, review, and maintenance of pump performance and test criteria. The team reviewed the in-service testing program to verify that ASME Code requirements were met. The team also conducted field walkdowns of portions of the pump, system piping, and associated supporting equipment to verify, by visual observation of reasonably accessible locations, that the installed configuration and material condition were consistent with the design bases and plant drawings.

b. Findings

Introduction: The team identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion XI, Test Control, for failure to correctly establish the acceptance criteria for the SDSW system.

Description: The licensee uses Procedure 34SV-P41-003-2, Standby Diesel Service Water System Operability, Version 4.5., to verify technical specification operability and IST requirements for the SDSW pump. Procedure 34SV-P41-003-2 established the lower limit of the pump differential pressure action range (0.93 of the reference value) as the acceptance criteria for the SDSW pump. The team determined that the licensee failed to consider minimum river level and maximum strainer differential pressure when establishing the acceptance criteria. The non-conservative acceptance criteria could result in the failure of the licensee to identify that the system was not capable of performing its design function during limiting system conditions. Based on the licensee's review of previous test procedure revisions, the non-conservative acceptance criteria existed since original plant operation.

The licensee initiated CR 2009105651 and performed a past operability determination. Additionally, the team reviewed the past 36 months of completed surveillances. The team's review determined that adequate margin existed for the SDSW pump during this period and thus no operability issue existed during the past 36 months.

The team also reviewed the licensee's evaluation of NRC Information Notice 97-90, Use of Non-conservative Acceptance Criteria in Safety-Related Pump Surveillance Tests, which addressed the concern described above. The team determined that this was a missed opportunity by the licensee to identify the non-conservative acceptance criteria.

Analysis: The licensee's failure to correctly establish acceptance criterion for the SDSW pump under the most limiting conditions was a performance deficiency. The team determined that the finding is greater than minor because it is similar to example 3.j of Appendix E, IMC 0612, in that it resulted in a condition where there was a reasonable doubt as to the operability of a component, and it adversely affected the Equipment Performance 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. Specifically, the failure to consider minimum river level and maximum strainer differential pressure when establishing the acceptance criteria resulted in reasonable doubt as to the capability of the SDSW pump. The team determined that the finding is of very low safety significance (i.e. Green) using the SDP because it did not represent a loss of system or safety function based on the team's review of the previous 36 months of surveillance data. The team determined that the finding does not represent current performance; therefore a cross-cutting aspect was not identified.

Enforcement: 10 CFR 50, Appendix B, Criterion XI, Test Control, states, in part, a test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Contrary to the above, since plant start-up in the 1970's, the licensee failed to incorporate the conservative acceptance limits into a procedure. Specifically, the licensee translated the non-conservative SDSW pump acceptance limits into Procedure 34SV-P41-003-2. Because this finding is of very low safety significance and because it was entered into the licensee's corrective action program as CR 2009105651, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000321/2009006-05 and 05000366/2009006-05, Failure to Correctly Establish Acceptance Criteria for the Standby Diesel Service Water Pump.

.2.9 Main Steam Line Pipe Whip Restraints (Units 1 and 2)

a. Inspection Scope

The team reviewed system description documents, safety analysis report, technical specifications and drawings to identify the design bases requirements of the main steam line pipe whip restraints. The team reviewed calculations for pipe stress analysis as well as evaluations performed in conjunction with the extended power uprate. Maintenance rule and in-service inspection documents were reviewed to ensure that the components were properly scoped. CRs were reviewed to identify adverse trends and to verify that potential degradation was monitored or prevented. The team also conducted field walkdowns of portions of the whip restraints to verify, by visual observation of reasonably accessible locations, that the installed configuration and material condition were consistent with the design bases and plant drawings.

b. Findings

Introduction: The team identified a Green non-cited violation of 10 CFR 50.65(a)(1) for the licensee's failure to monitor in accordance with established criteria the main steam line and feedwater line pipe whip restraints for Units 1 and 2.

Description: The Unit 1 and 2 main steam and feedwater line pipe whip restraints are passive components designed to protect safety-related components from damage due to pipe rupture caused by a high energy line break. The whip restraints were scoped into the licensee's Maintenance Rule Program in 1996.

The team noted that since plant start-up the licensee failed to inspect the restraints on Units 1 and 2. The team determined that the licensee lacked reasonable assurance that the restraints were capable of fulfilling their intended function because the restraints had not been inspected over the life of the plant. During the inspection, the licensee performed visual inspections of accessible restraints and did not identify any deficiencies. The completion of the inspections was complicated by accessibility and by the site's lack of design drawings for the Unit 1 restraints. The licensee initiated CRs 2009105147 and 200910622, and plans to complete the inspections during the upcoming Unit 1 and 2 outages.

Analysis: The team determined that the licensee's failure to periodically inspect the condition of the safety-related pipe whip restraints was a performance deficiency. This finding is more than minor because it is associated with equipment performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to perform periodic inspections or preventative maintenance resulted in a lack of reasonable assurance that the restraints would perform their safety function. The team determined that the finding is of very low safety significance (Green) using the SDP because the finding did not represent an actual loss of safety function. The finding directly involved the cross-cutting aspect of implementing a corrective action program with a low threshold for identifying issues under the Corrective Action Program (CAP) component of the Problem Identification and Resolution area. Specifically, the licensee failed to enter the lack of inspections into the CAP: when it was identified that Unit 1 pipe restraint design documents could not be located in 1997; and, when the restraints were added to the structural monitoring program for license renewal in 2009 [P.1(a)].

Enforcement: 10 CFR 50.65(a)(1) states, in part, that licensee's shall monitor the performance or condition of structures, systems and components (SSCs) within the scope of the rule as defined by 10 CFR 50.65 (b), against license established goals, in a manner sufficient to provide reasonable assurance that such SSCs are capable of fulfilling their intended function.

10 CFR 50.65(a)(2) states, in part, that monitoring as specified in (a)(1) is not required where it has been demonstrated that the performance or condition of a component is being effectively controlled through the performance of appropriate preventive maintenance such that the SSC remains capable of fulfilling its intended function. Contrary to the above, the licensee failed to demonstrate that the performance or condition of the pipe whip restraints had been effectively controlled through the performance of appropriate preventive maintenance and did not monitor against licensee established goals. Specifically, since 1996 the licensee failed to perform inspections or preventative maintenance for the main steam and feedwater pipe whip restraints. Because this finding is of very low safety significance and because it was entered into the licensee's corrective action program as CR 2009105147 and 200910622, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000321/2009006-06 and 05000366/2009006-06, Failure to Monitor the Main Steam and Feedwater Line Pipe Whip Restraints.

.2.10 Feedwater Pipe Whip Restraints (Units 1 and 2)

a. Inspection Scope

The team reviewed system description documents, safety analysis report, technical specifications and drawings to identify the design bases requirements of the feedwater line pipe whip restraints. The team reviewed calculations for pipe stress analysis as well as evaluations performed in conjunction with the extended power uprate. Maintenance rule and in-service inspection documents were reviewed to ensure that the components were properly scoped. CRs were reviewed to identify adverse trends and to verify that potential degradation was monitored or prevented. The team also conducted field walkdowns of portions of the whip restraints to verify, by visual observation of reasonably accessible locations, that the installed configuration and material condition were consistent with the design bases and plant drawings.

b. Findings

See Section 1R21.2.9

.2.11 4.16 kV Emergency Bus 2G Inspection Scope

a. Inspection Scope

The team reviewed AC load flow calculations to verify that the 4160V system had sufficient capacity to support its required loads under worst case accident loading and grid voltage conditions. The team reviewed the design of the 4160V bus degraded voltage protection scheme to determine whether it afforded adequate voltage to safety related devices at all voltage distribution levels. This included review of degraded voltage relay setpoint calculations, alarm response procedures, motor starting and running voltage calculations, and motor control center (MCC) control circuit voltage drop calculations. The team reviewed maintenance procedures to determine whether they included up to date vendor technical data. The team reviewed maintenance procedures, schedules, and vendor recommendations for the 4160V switchgear and its associated circuit breakers to determine whether equipment was being properly maintained. The team reviewed corrective action documents and maintenance records to determine whether there were any adverse operating trends. The team reviewed bus transfer schemes and associated calculations to determine whether automatic transfer functions could be performed without degrading safety related equipment. The team reviewed operating procedures to determine whether the limits and protocols for maintaining offsite voltage were consistent with design calculations. In addition, the team performed a walkdown of the 4160V safety buses to assess operability and condition.

b. Findings

Introduction: The team identified an Unresolved Item (URI) regarding the licensee's calculation that evaluated the adverse effects of a postulated early transfer of non-safety buses to a start-up transformer that energizes safety-related buses.

Description: Unit 2 utilizes three safety-related buses (2E, 2F, and 2G) to energize emergency core cooling system equipment. Safety-related buses 2F and 2G are normally connected to the same winding of start-up transformer 2D. Hatch utilizes a fast transfer scheme that allows the same winding of the start-up transformer 2D to energize non-safety bus 2D following a main generator trip. The fast transfer results in non-safety bus 2D and safety-related buses 2F and 2G being energized from the same winding of start-up transformer 2D.

For a main generator trip that's caused by high drywell pressure or low reactor water level, the fast transfer of non-safety bus 2D occurs several seconds after the large ECCS motors have block started onto the safety-related buses 2F and 2G. If a single failure is postulated, the fast transfer would be concurrent with ECCS block motor starting. A postulated single failure could be caused by the faults to the following components: UAT relaying, normal supply breaker circuits, or main generator tripping scheme. The effect of the fast transfer occurring with a postulated single failure is that the electrical power for two trains of safety-related equipment would be adversely affected.

The licensee analyzed the vulnerability described above in Calculation SENH 92-133, Bus Transfer Study, Rev. 1 and determined that if a failure as described above occurred, safety-related bus voltage would dip to approximately 48% of 4160V for 1.09 seconds during motor starting. Calculation SENH 92-133 showed that large motors would start, but the team had additional questions regarding modeling techniques used for systems experiencing voltage dips greater than 30%. In addition, the team had questions regarding why the calculation did not evaluate the effect of the voltage dip on other safety related equipment connected to the system.

Summary: This item is unresolved pending the NRC's review of Calculation SENH 92-133 to determine the adequacy of the methodology, and the NRC's review to determine if the postulated scenario is within the Hatch licensing bases. The issue is applicable to both Units 1 and 2. (URI 05000321/2009006-07 and 05000366/2009006-07, Postulated Early Transfer of Non-Safety Buses)

.2.12 600VAC MCC 2R24-S012

a. Inspection Scope:

The team reviewed AC load flow calculations to verify that the 600V bus had sufficient capacity to support its required loads under worst case accident loading and grid voltage conditions. The team reviewed the degraded voltage protection scheme to verify that the voltage setpoints were selected based on the voltage requirements for safety related loads at the 600V level. The team reviewed Annunciator Response Procedures to determine whether response to 600V system undervoltage was appropriate. The team reviewed system health reports, corrective action documents, and maintenance records to determine whether there were any adverse operating trends. In addition, the team performed a walkdown of the 600V safety buses to assess operability and condition. Additionally, the team reviewed FSAR Supplement 15A, system description documents, equipment qualification reports and calculations, and drawings to identify the equipment qualification requirements for a high energy line break. The team selectively reviewed maintenance work orders, purchase orders and work orders to confirm that the equipment qualification was maintained in accordance with the design bases. In addition, the team selectively reviewed design change packages associated with the MCC to verify that the design change activity did not affect the original equipment qualification of the MCCs or equipment contained therein. The team also performed non-intrusive visual inspections of the motor control center to assess visible material condition and vulnerability to hazards (flooding, seismic interactions, and missiles).

b. Findings

Introduction: The team identified an Unresolved Item (URI) regarding the Hatch degraded voltage protection scheme. The existing automatic degraded voltage protection scheme employs automatic setpoints that are too low to assure operability of safety related electrical equipment in case of a sustained degraded grid condition, and instead relies on administrative controls to assure adequate voltage to safety-related equipment during an accident.

Description: In 1991, the NRC Engineering Design Safety Function Inspection determined that Hatch's calculations for the setpoints of the inverse time degraded undervoltage protection relays, then set at approximately 78.8% with a 20 second delay, were not adequate. Hatch updated the voltage calculations, and indicated in a letter dated November 22, 1993 that the setpoints would need to be raised to approximately 91% of 4160V at the 4160V safety buses in order to ensure adequate voltage to safety related loads during a LOCA. Graphs attached to the letter showed that required LOCA voltages ranged from 88.46% to 90.8% for the three 4160V safety buses. During the inspection, Hatch was not able to locate calculations that supported the values (88.46% to 90.8%) given in the graph.

Hatch concluded that raising the trip setpoint to 91% would result in little margin between the trip setpoint at which the buses would be separated from offsite power, and the minimum bus voltage that could occur if offsite declined to the lower end of its expected range (101.3% of 230kV). Because of the increase in risk of spurious separation of the offsite power supply that would have occurred if the trip setting of the undervoltage relay was raised, Hatch proposed a scheme where the trip setpoint of the relays providing the automatic separation feature would remain at its existing setting, and additional relays providing an alarm function would be installed, with a setpoint of approximately 92%. In addition, Hatch agreed to maintain a minimum switchyard voltage of 101.3% of 230kV, supported by a software based contingency alarm operated by the transmission system operator. This scheme was recognized as a deviation from the guidance on degraded voltage protection provided in NRC generic letter dated June 2, 1977 because it relied on an alarm followed by manual operator action, in lieu of automatic protection, but it was accepted by the NRC in an SER dated February 23, 1995. Consequently, Hatch is currently relying on measures implemented and maintained by their transmission system operator to assure adequate power to safety related equipment during an accident.

Summary: This item is unresolved pending further NRC review of plant design and prior NRC inspections related to this issue. (URI 05000321/2009006-08 and 05000366/2009006-08, Degraded Voltage Protection)

.2.13 RWCU Annunciators and Associated Functions

a. Inspection Scope

The team reviewed instrument loop diagrams and wiring diagrams to qualification documentation was consistent with actual location of trip circuits. The team reviewed setpoint calculations and associated procedures verify that they are consistent with each other. The team reviewed completed surveillances to determine whether equipment performance was consistent with Technical Specifications, and whether anomalies were addressed by the corrective action program. The team reviewed environmental conditions to and calculations verify that setpoints have sufficient margin to account for environmental drift effects. The team reviewed the modification for the replacement of Barton transmitters with qualified Rosemount transmitters.

b. Findings

No findings of significance were identified.

.2.14 HPCI Annunciators and Associated Functions

a. Inspection Scope

The team reviewed the adequacy of instrumentation and controls for closing the HPCI turbine inboard/outboard steam isolation valves, and torus suction valves. The team reviewed instrument loop diagrams and wiring diagrams to verify that instrumentation and control schemes for high energy line break isolation functions were consistent with the design bases. The team reviewed instrument setpoint calculations, calibration procedures, and surveillance records verify that instrument setpoints were adequate and were being maintained consistent with the Technical Specifications, and whether anomalies were addressed by the corrective action program. The team reviewed environmental conditions and calculations to determine whether setpoints have sufficient margin to account for environmental drift effects. The team examined the instruments and their environs with remote video to assess installed configurations and materiel condition.

b. Findings

No findings of significance were identified.

.2.15 125/250 VDC Motor Control Center (2R24-S022)

a. Inspection Scope

The team reviewed FSAR Supplement 15A, system description documents, equipment qualification reports and calculations, and drawings to identify the equipment qualification requirements for a high energy line break. The team selectively reviewed maintenance work orders, purchase orders and work orders to confirm that the equipment qualification was maintained in accordance with the design bases. In addition, the team selectively reviewed design change packages associated with the MCCs to verify that the design change activity did not affect the original equipment qualification of the MCC's or equipment contained therein. The team also performed non-intrusive visual inspections of the MCC to assess visible material condition and vulnerability to hazards (flooding, seismic interactions, and missiles)

b. Findings

No findings of significance were identified.

2.16 Reactor Building Floor and Equipment Drainage System Annunciators and Level Switches (Units 1 and 2)

a. Inspection Scope

The team reviewed design criteria documents, system description documents, technical specifications, and drawings to identify the design bases functions. The team reviewed functional test data, calibration data and valve actuation data to verify the design function of the level switches and annunciation functions. The team also performed a walkdown of the diagonal rooms to assess the observable material condition of the level switches, vulnerability to hazards, and effectiveness of design features.

b. Findings
See Section 1R21.2.9

.3 Review of Low Margin Operator Actions

a. Inspection Scope

The team performed a margin assessment and detailed review of five risk significant and time critical operator actions. Where possible, margins were determined by the review of the assumed design basis and UFSAR response times. For the selected operator actions, the team performed a walkthrough of associated Emergency Operating Procedures (EOPs) Abnormal Operating Procedures (AOPs), Annunciator Response Procedures (ARPs), and other operations procedures with plant operators and engineers to assess operator knowledge level, adequacy of procedures, availability of special equipment when required, and the conditions under which the procedures would be performed. Detailed reviews were also conducted with operations and training department leadership, and through observation and utilization of a simulator training period to further understand and assess the procedural rationale and approach to meeting the design basis and UFSAR response and performance requirements. Operator actions were observed on the plant simulator and during plant walk downs. Operator actions associated with the following events/evolutions were reviewed:

- Operator Response to Primary System Coolant Pipe Break in Reactor Building
- Operator Response for Secondary Containment Control
- Operator Actions for Station Blackout
- Operator Response to Loss of Room Cooling
- Operator Response to Loss of Instrument air

b. Findings

No findings of significance were identified.

.1 Review of Industry Operating Experience

a. Inspection Scope

The team reviewed selected operating experience issues that had occurred at domestic and foreign nuclear facilities for applicability at the Edwin I. Hatch Nuclear Plant. The team performed an independent applicability review for issues that were identified as applicable to the Edwin I. Hatch Nuclear Plant and were selected for a detailed review. The issues that received a detailed review by the team included:
IN 2006-15, Vibration-Induced Degradation and Failure of Safety-Related Valves
IN 2006-23, Vibration-Induced Degradation of Butterfly Valves
IN 2002-12, Submerged Safety-Related Electrical Cables

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA6 Meetings, Including Exit

On June 5, 2009, the team presented the inspection results to Mr. Bargeron and other members of the licensee's staff. No proprietary information was reviewed as part of this inspection.

Additionally, a re-exit was performed on July 21, 2009 and the status of unresolved items was presented to Mr. Bargeron and other members of the licensee's staff.

ATTACHMENT: SUPPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:
S. Tipps, Licensing Principal Engineer
D. Hine+s, Site Design Engineering Supervisor
H. Nipper, Engineering Support Principal Engineer

NRC personnel
B. Desai, Chief, Engineering Branch Chief 1, Division of Reactor Safety, RII
C. Christensen, Deputy Director, Division of Reactor Safety, RII
J. Hickey, Senior Resident Inspector, Hatch
P. Niebaum, Resident Inspector, Hatch

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000321/2009006-01 and 05000366/2009006-01	NCV	Failure to Perform Cause Determinations and Corrective Actions for Deficiencies in Containment Penetration Seals (Section 1R21.2.1)
05000366/2009006-02	NCV	Failure to Correctly Establish Containment Isolation Valve Leakage Criteria for the Unit 2 Feedwater Check Valves (Section 1R21.2.2)
05000321/2009006-05 and 05000366/2009006-05	NCV	Failure to Correctly Establish Acceptance Criteria for the Standby Diesel Service Water Pump Section (Section 1R21.2.8)
05000321/2009006-06 and 05000366/2009006-06	NCV	Failure to Monitor the Main Steam and Feedwater Line Pipe Whip Restraints (Section 1R21.2.9)

Opened

05000321/2009006-03 and 05000366/2009006-03	URI	Non-Interruptible Essential Instrument Air Header Check Valves for Units 1 and 2 (Section 1R21.2.6)
05000321/2009006-04 and 05000366/2009006-04	URI	Reactor Building Equipment Drain Sump System for Units 1 and 2 (Section 1R21.2.7)
05000321/2009006-07 and 05000366/2009006-07	URI	Postulated Early Transfer of Non-Safety Buses (Section 1R21.2.11)
05000321/2009006-08 and 05000366/2009006-08	URI	Degraded Voltage Protection (Section 1R21.2.12)

LIST OF DOCUMENTS REVIEWED

Calculations

0071 (V999, B999), Standby Service Water System Flow Model, Rev. 0
532-40-6511, Temperature Setpoint Leak Detection, Rev. 0
560 (V3, B43), Evaluate the Temperature Response for the RHR Corner Room, Rev. 1
BH1-M-V004-B007-0001, Compartment Long Term Temperature Analysis, Ver. 9
SMNH-05-014, Post LOCA Time Elapse, Ver. 1
SMNH-09-004, Inservice Test Program Allowable Leakage Rate, Ver. 1, 2 and 3
BH1-CS-33-T48-13, Pressure Switch K626A Setpoint Determination, Rev. 1
BH1-M-V001-0005, Water Level of Torus Water in Basement, Rev. 0
BH1-M-V005-0185, RWCU Line Break Max Flood in the NE and SE Diagonals, Rev. 0
BH1-M-V005-0182, RWCU Line Break Flow Vs Time and Max Flow, Rev. 0
BH1-M-V005-0183, RWCU Line Break Flood Elevations Vs Time, 158' Elevation, Rev. 0
BH1-M-V005-0184, Flood Height Vs Flood Volume in East Corner Rooms, Rev. 0
BH2-CS-33-2P52-01, Pressure Switch K626A Setpoint Determination, Rev. 1
BH2-M-0387, Torus Room Flood Level Due to Feedwater Line Break, Rev. 0
BH2-M-0585, RWCU Line Break, 158' Elevation, Rev. 0
BH2-M-0586, Flood Height Vs Flood Volume in East Corner Rooms, Rev. 0
BH2-M-0587, RWCU Line Break Max Flood in the NE and SE Diagonals, Rev. 0
BH2-M-0630, Control Building Heat Up Analysis for IPE, Rev. 1
BH2-M-V999-0071, Standby Service Water System Flow Model, 09/10/2002
SMNH-08-011, NPSH and Minimum Submergence Requirements, 10/31/2007
0001 Vol. 999 Subject 08, – Evaluation of Pipe Whip Restraints, Rev. 1
Calculation BH1-PD-2716, Pipe Stress Analysis of Main Steam Lines A, B, C & D, Rev. 7
BH1-M-V004-B007-0001, Compartment Long Term Temperature Analysis, 6/8/04
342 (V3, B 26), Torus Room Environment due to a MSLB, Rev. 3
86053MP, Hatch Unit 1 Data Base Update, Rev. 1
SENH 03-006, Station Auxiliary System Study (Unit 1), Rev. 3
SENH 03-007, Station Auxiliary System Study (Unit 2), Rev. 2
SENH 04-001, Degraded Grid (Diesel Generator) Alarm Relay Setpoint, Rev. 3.0
SENH 04-002, Degraded Grid (Diesel Generator) Alarm Relay Setpoint, Rev. 3.0
SENH 92-133, Bus Transfer Study, Rev. 1
SENH 95-005, Evaluate Class 1E Station Auxiliary System, Rev. 5
SENH 96-005, Evaluate Class 1E Station Auxiliary System (Unit 1), Rev. 5
SENH 96-006, Evaluate Class 1E Station Auxiliary System (Unit 2), Rev. 4
SENH 97-003, Unit 2 As-Built Base Calculation for Safety Related AC MOV's, Rev. 2
SINH 01-046, Technical Specification 3.3.6.1-1 (3.i) Setpoint Determinations, Rev. 0
SINH 01-053, Technical Specification 3.3.6.1-1 (5a) Setpoint Determinations, Rev. 0

Procedures

34SV-P41-003-2, Standby Service Water System Operability, Ver. 4.5
42IT-TET-014-1, Safeguards Equipment Room Cooler Performance Test, Ver. 2
42SV-TET-001-1, Primary Containment Periodic Type B and Type C Leakage Tests, Ver. 24.2
42SV-TET-001-2, Primary Containment Periodic Type B and Type C Leakage Tests, Ver. 30.1
31EO-EOP-012-1, Primary Containment Control, Ver. 5
31EO-EOP-012-2, Primary Containment Control, Ver. 5
31EO-EOP-014-1, Secondary Containment Control, Ver. 8
31EO-EOP-014-2, Secondary Containment Control, Ver. 7
31EO-EOP-101-2, Emergency Containment Venting, Ver. 4.2

34AB-P51-001-1, Loss of Instrument and Service Air System, Ver. 4.4
34AB-R22-002-1, Loss of 4160V Emergency Bus, Ver. 1.4
34AB-R22-003-1, Station Blackout, Ver. 3.5
34AB-R22-002-2, Loss of 4160V Emergency Bus, Ver. 1.8
34AB-R22-003-2, Station Blackout, Ver. 3.6
34AB-R43-001-1, Diesel Generator Recovery, Ver. 1.12
34AB-R43-001-2, Diesel Generator Recovery, Ver. 1.14
34AB-T22-002-1, Loss of Secondary Containment Integrity, Ver. 0.4
34AB-T22-003-1, Secondary Containment Control, Ver. 5.7
34AB-T22-002-2, Loss of Secondary Containment Integrity, Ver. 1.1
34AB-T22-003-2, Secondary Containment Control, Ver. 3.8
34AB-T41-001-1, Loss of ECCS, MCREC, or Area Ventilation System(s), Ver. 3.6
34AB-T41-001-2, Loss of ECCS, MCREC, or Area Ventilation System(s), Ver. 3.5
34AR-657-901-2, ARPs for Control Panel 2H11-P657, Alarm Panel 1, Ver. 22.12
34SO-P70-001-1, Drywell Pneumatic System, Ver. 11.4
34SO-P70-001-2, Drywell Pneumatic System, Ver. 10.4
34SO-T41-001-1, Core Spray and RHR Rooms Ventilation System, Ver. 4.5
34SO-T41-001-2, Core Spray and RHR Rooms Ventilation System, Ver. 3.3
64CH-SAM-018-0, Instrument Air Sampling, Ver. 2.4
34SV-P41-003-2, Standby Diesel Service Water Operability, Version 4.5
52SV-R43-001-0, Diesel, Alternator, and Accessories Inspection, Version 20.3
34IT-T45-001-1, Reactor Building Instrument Sumps Isolation Valve Exercise, Ver. 6
34IT-T45-001-2, Reactor Building Instrument Sumps Isolation Valve Exercise, Ver. 4
57CP-T45-002-1, GEMS LS 800 Level Switch Calibration, Rev. 4
57CP-T45-002-1, GEMS LS 800 Level Switch Calibration, Rev. 4.1
57CP-CAL-094-2, Robert Shaw Level Switches, Rev. 10.20
34IT-T45-001-1, Reactor Building Instrument Sumps Isolation Valves Exercise, Rev. 0.6
34IT-T45-001-2, Reactor Building Instrument Sumps Isolation Valves Exercise, Rev. 0.4
57IT-T45-002-1, Sump Isolation Valve Actuation Test, Rev. 1.1
57IT-T45-002-1, Sump Isolation Valve Actuation Test, Rev. 1.2
57IT-T45-002-2, Sump Isolation Valve Actuation Test, Rev. 0.2
34AR-657-034-1, R/B N-E Diagonal Floor Drain Sump Level High-High, Rev. 3.1
34AR-657-016-1, R/B N-E Diagonal Floor Drain Sump Level High, Rev. 6
34AR-657-033-1, R/B N-W Diagonal Floor Drain Sump Level High, Rev. 6.1
34AR-657-016-2, R/B S-E Diagonal Floor Drain Sump Level High, Rev. 6.1
34AR-657-033-2, R/B S-W Diagonal Floor Drain Sump Level High, Rev. 4.2
34AR-657-034-2, R/B S-E Diagonal Floor Drain Sump Level High-High, Rev. 4.2
34AB-R23-001-1, Loss of 600V Emergency Bus, Rev. 1.1
34AB-R23-001-2, Loss of 600V Emergency Bus, Rev. 1.4
34AB-S11-001-0, Operation with Degraded System Voltage, Rev. 2.3
34AR-651-901-2, ARPs for Control Panel 2H11-P651, Alarm Panel 1, Rev. 14.6
34AR-652-901-1, ARPs for Control Panel 1H11-P642, Alarm Panel 1, Rev. 11.8
34SV-SUV-019-2, Surveillance Checks, Rev. 34.20
52IT-R22-001-2, Fast Transfer Time Testing of BOP 4160V ACBs, Rev. 2.6
52PM-R22-001-0, Westinghouse 4160 VAC Switchgear and Components P.M., Rev. 24.9
52PM-Y46-001-0, Inground Pullbox and Cable Duct Inspection for Water, Rev. 6.9
57CP-CAL-097-1, Rosemount 1153 and 1154 Transmitters, Rev. 24.11
57GM-MIC-006-0, Removal and Installation of GE/WEED RTDs, Rev. 1.9
57SV-CAL-004-1, ATTS RTD Calibration, Rev. 6.5

65h-020 37, Maximum Allowable AC Motor Starter Control Circuit Length, Rev. 3

Completed Procedures
34SV-P41-003-2, Standby Service Water System Operability, 9/18/2008
42SV-TET-001-2, Summarized LLRT Test Results, 12/3/2007
34SV-P41-003-2, Standby Diesel Service Water Operability, 7/29/2003 - 3/19/2009
34SV-R43-012-2, Diesel Generator 1B 24 Month Operability Test, 7/18/2008
64CH-SAM-018-0, Instrument Air Sampling, 03/24/2009
34IT-T45-001-1, Reactor Building Instrument Sumps Isolation Valve Exercise, 4/20/2009
34IT-T45-001-2, Reactor Building Instrument Sumps Isolation Valve Exercise, 4/21/2009
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2002102813, Breaker Would Not Close In With Handle Mounted on Door
2005102276, Valve Did Not Have Closed Light Indication; Valve and Limit Switches Found Underwater
2005103376, Instrument Sump Isolation Valve Indication Lights Not Operating Properly
2005109322, 1E41-N670A Out of Tolerance
2006101120, 1T45F004, RHR S-E Outboard Sump Isolation Valve Would Not Close
2006101120, Valve Would Not Close When Control Switch Was Placed In Closed Position
2006105410, 1T45F004 Would Not Stroke When Tested, 05/15/2006
2006105410, Valve Would Not Stroke When Tested
2006106459, No Valve Position Indication with Control Switch in Auto, 06/19/2006
2006109385, Functional Failure of the Standby Plant Service Water Pump, 09/15/2006
2006109879, Did Not Stroke Fully Closed When Control Switch Was Placed in Closed Position
2007101560, Valve Has Double Indication with Control Switch in Auto, 2/11/2007
2007101727, Valve Was Not Stroking All The Way Due to Air Operator Having Water in Cylinder
2007101771, Six Of Eight Unit 2 MSIVs Failed their LLRT
2007102246, Operating Tab for Mechanical Interlock Broken Off
2007102351, Seismic Screws Found Missing
2007104911, Valve Exercise Results Unsat; Valve Does Not Give a Full Closed Indication
2007107211, 4-Way Solenoid Valve Installed Instead Of 3-Way Solenoid Valve; Plugging One Port Could Cause Spool to Hydraulically Lock

2007107211, Discovered Two Air Lines to the Operating Cylinder on 2T45-F006
2007110335, 1E41-N670A Out Of Tolerance
2007110336, 1E41-N671A Out Of Tolerance
2007110479, Valve Has Double Indication with Control Switch in Auto and Closed Positions
2007110605, Correction of Technical Specifications for Unit 1 Alignment, 11/27/2007
2007111120, Valve Had Double Indication with Switch in Auto; Indicates Closed After Cycling From Closed To Auto
2008100804, Breaker Failed Test – Replacement Installed
2008100816, Valve Indicated Closed Prior To Valve Exercise and Had Closed Indication After The Exercise
2008101840, Atts Instrument 2G31-N662D Drifting
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2008102508, Double Indication Failure during Ops Surveillance Testing
2008104838, Selected Level Switches Not Calibrated In Last Four Years
2008107664, Valve Has No Position Indication
2008107690, Valve Has an Observable Double Indication In Control Room; It Cannot Be Determined If The Valve Is Open or Closed
2008109636, 2T45-F002 Has a History of Valve Position Indication Problems
2008111851, Records Update For Barton Transmitter Replacement
2009101098, Unit 2 Drywell Penetration 2T52-X105A Exceeded Maximum Allowable Leakage
2009101102, Unit 2 Drywell Penetration 2T52-X105C Exceeded Maximum Allowable Leakage
2009101165, Penetration 2T23-X42 Exceeded LLRT Administrative Limits
2009101202, Penetration 2T23-X11 Exceeded LLRT Administrative Limits
2009101206, Penetration 2T23-X42 Exceeded LLRT Administrative Limits
2009101209, Penetration 2T23-X12 Exceeded LLRT Administrative Limits
2009101243, Penetration 2T23-X10 Exceeded LLRT Administrative Limits
2009101245, Penetration 2T23-X7A Exceeded LLRT Administrative Limits
2009101247, Penetration 2T23-X9B Exceeded LLRT Administrative Limits
2009101249, Penetration 2T23-X8 Exceeded LLRT Administrative Limits
2009101411, Electrical and Mechanical Penetration Exceeded LLRT Administrative Limits
2009105572, 2P41F341 Does Have a Safety Function to Close, 06/01/2009
2009105651, Surveillance Acceptance Limits Based On ASME IST Testing, 06/02/2009

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2006201277, Revise Procedure to Include Positive Retention of Tapered Pins in Butterfly Valves, 10/31/2006
2007204027, More Time is Needed to Determine What Should Be Installed, 09/26/2007
2008203661, Investigate Water Being Found and Why It Is Occurring So Frequently, 05/29/2009
2009200612, Engineering Needs To Identify the Appropriate 3-Way Solenoid Valve To Be Installed, 02/10/2009
2009200613, Replace 4-Way Solenoid Valves On 2T45F001-F007 with 3-Way Solenoid Valves, 02/10/2009
2009200784, If Required, Generate an ED For The Replacement Of The 2T45F001-F007 4-Way Solenoid Valves To 3-Way Solenoid Valves 02/25/2009

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1060288801, 1T45F004 RHR S-E Outboard Sump Isolation Valve, 02/06/2006
1082340701, Reactor Building Instrument Sumps Isolation Valves Exercise, 4/20/2009
2000086801, Replace Breaker Springs In MCC, 3/21/2001
2010200901, Bowed Relay, Repair/Replace, 7/31/2001
2040713301, 2T45F006 Has an Air Leak At The Piston Rod Seal Area, 03/25/2004
2040902701, Perform Circuit Breaker Inspection/Test and MCC Minor Inspection, 2/27/2007
2041520901, Robert Shaw Level Switch Calibration for 2T45-N006, 5/19/2005
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2050710301, 2T45-F003 Did Not Have a Close Light Indication, 03/13/2005
2050710301, Adjust Limit Switches of Valve 2T45-F003 in Order to Get Proper Indications, 6/16/2005
2051399807, Replace HFB Circuit Breakers with HFD Circuit Breakers In MCC, 2/21/2007
2051399809, Replace HFD Breakers with HFD Breakers And Re-pull Cable from MCC To Motor Starter, 5/10/2006
2052521501, Replace MCC Circuit Breaker, 2/1/2007
2061435101, Valve 2T45-F002 Has No Valve Position Indication, 06/19/2006
2061493401, Light Indication for All Instrument Sump Isolation Valves Are Extinguished, 07/01/2006
2070429901, Sump Isolation Valve Has Double Indication, 02/11/2007
2070475301, Found That Valve Was Not Stroking All The Way, 02/14/2007
2070536201, Replace Overload Heaters, 2/28/2007
2070617001, Replace Breaker Closing Mechanism, 9/21/2007
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2090320901, Replace The 4-Way Solenoid Valves On The 2T45F001-F007, 02/10/2009,
2090320902, Replace The 4-Way Solenoid Valves On The 2T45F001, 02/11/2009
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2980172101, Replace Heater Elements on Allis Chalmers MCC, 9/9/1998
2980172101, Replace Overload Relay and Heaters, 9/10/1998
2980194501, Replace Terminal Board In MCC, 10/8/1998
2989500914, Replace MCC Circuit Breaker, 10/30/1998

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2009104567, Allowable Leakage for Unit 2 Feedwater Check Valves
2009104682, Correct the micron size description in the FSAR
2009104683, Drawing H16329 was reviewed and found to be in error
2009105109, Non-interruptable instrument air check valves not monitored in maintenance rule
2009105110, Unit 1 level switches are not being functionally tested
2009105111, Level switches in torus instrument sumps and the level switches/isolation valves associated with the ECCS diagonal sumps are not currently scoped in maintenance rule
2009105124, CR 2008101941 performance tab for maintenance rule was incorrect
2009105130, Previous Calculation of Record not attached to current calculation
2009105132, Cotton glove stuck to small electrical conduit in Unit 2 reactor building
2009105137, Dewpoint analyzer not responding to process dewpoint temperature changes
2009105139, FSAR notes peak accident temperature of 214 degrees while most EQ documents note a peak accident temperature of 213 degrees
2009105147 Pipe Whip Restraints Not Inspected Over the Life of the Plant 05/20/2009
2009105209, Two 2" instrument air check valves found without labels
2009105377, Clutter in Unit 1 RWCU heat exchanger room
2009105427, Administrative Controls Needed to Comply with Station Blackout Safety Evaluation Report Recommendations
2009105476, Pipe stress Calculation BH1-PD-2691 discovered in file with partially completed revision
2009105572, Stand-by Service Water Air release Valve Not in IST Program
2009105615, A single point vulnerability exists with the reactor building sump level instrumentation
2009105622 Lack of Drawings For Unit 1 Pipe Whip Restraints
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2009105727, While performing 57IT-T45-002-1 an unexpected alarm occurred
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