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

OCTOBER 24, 2002

Southern Nuclear Operating Company, Inc.
ATTN: Mr. H. L. Sumner, Jr.
Vice President - Hatch Plant
P. O. Box 1295
Birmingham, AL 35201-1295

SUBJECT: EDWIN I. HATCH NUCLEAR POWER PLANT - NRC INSPECTION REPORT 50-321/02-06, 50-366/02-06

Dear Mr. Sumner:

On September 13, 2002, the Nuclear Regulatory Commission (NRC) completed a safety system design and performance capability inspection at your E. I. Hatch facility, Units 1 and 2. The enclosed report documents the inspection findings which were discussed with Mr. Pete Wells and other members of your staff during an exit meeting on September 13, 2002, and with Mr. Steve Tipps and other members of your staff during a telephone conference on October 18, 2002.

The inspection was an examination of 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 operating license. Within these areas, the inspection involved selected examination of procedures and representative records, observations of activities, and interviews with personnel.

On the basis of the sample selected for review, the design, performance capability, and procedures for equipment used to mitigate the “Loss of Station Battery A” event were properly controlled. There were two green findings identified during this inspection. These findings were determined to be violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these findings as non-cited violations, in accordance with Section VI.A.1 of the NRC’s Enforcement Policy. If you deny either of these non-cited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the U. S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-001, with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-001; and the NRC Resident Inspector at your E. I. Hatch facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Charles R. Ogle, Chief
Engineering Branch 1
Division of Reactor Safety

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

Enclosure: NRC Inspection Report 50-321/02-06, 50-366/02-06 w/Attachment

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E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

U. S. NUCLEAR REGULATORY COMMISSION
REGION II

Docket Nos.:	50-321, 50-366
License Nos.:	DPR-57, NPF-5
Report Nos.:	50-321/02-06, 50-366/02-06
Licensee:	Southern Nuclear Operating Company, Inc. (SNC)
Facility:	E. I. Hatch Nuclear Power Plant, Units 1 & 2
Location:	P.O. Box 2010 Baxley, Georgia 31515
Dates:	August 26-30, 2002 and September 9-13, 2002
Inspectors	R. Schin, Senior Reactor Inspector (Lead Inspector) P. Fillion, Reactor Inspector S. Freeman, Resident Inspector, Oconee M. Maymi, Reactor Inspector (Week 1 only) N. Merriweather, Senior Reactor Inspector
Accompanied by:	Y. Diaz, Inspector Trainee R. Fanner, Inspector Trainee J. Moorman, Team Leader C. Ogle, Chief, Engineering Branch 1 R. Taylor, Inspector Trainee
Approved by:	C. Ogle, Chief Engineering Branch 1 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000321-02-06, IR 05000366-02-06; Southern Nuclear Operating Company, Inc.; on 08/26/2002 - 09/13/2002; E. I. Hatch Nuclear Power Plant, Units 1 & 2; Safety System Design and Performance Capability Inspection.

This inspection was conducted by a team of regional and resident inspectors. Two green findings with related non-cited violations were identified during this inspection. The significance of the findings is indicted 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 Oversite Process," Revision 3, dated July 2000.

<u>Inspector Identified Findings</u>

Cornerstone: Mitigating Systems

- Green. A non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control, was identified for inadequate design control of the high pressure coolant injection (HPCI) system suction source from the condensate storage tank (CST). Vortexing in the CST was not accounted for when the licensee calculated the CST level setpoint specified in the Technical Specifications (TS) for automatic HPCI system suction switchover from the CST to the suppression pool. Vortexing could cause air ingestion into the HPCI system suction from the CST and the air could then damage the HPCI pump.

  This finding was of very low safety significance because licensee use of the non-safety CST as a HPCI pump suction source with the CST at low levels was unlikely since the reactor vessel or suppression pool would generally reach a high level first, where the HPCI pump would be automatically stopped or its suction would be automatically switched to the safety-related suppression pool. In addition, alternate core cooling methods would normally be available, including reactor core isolation cooling (RCIC) as well as automatic depressurization system (ADS) and low pressure coolant injection (LPCI). (Section 1R21.1 b (1))

- Green. A non-cited violation of TS 5.4.1, Procedures, was identified for an inadequate RCIC system operating procedure. The section of the procedure for local manual operation of RCIC, if followed exactly as written, would have resulted in overspeeding the RCIC pump with no water flow through the pump and with no cooling water to the pump.

  This finding was of very low safety significance because the likelihood of losing Division I direct current (DC) power was low; consequently the potential need for local manual operation of RCIC was low. In addition, other core cooling methods would normally be available, including HPCI as well as ADS and LPCI.
  (Section 1R21.1 b. (2))

- TBD. An unresolved item was identified. The steam line drains on the steam supply line to the Division II HPCI system could fail to function as a result of a failure of the Division I DC power system. This could allow water intrusion into the HPCI turbine and possibly damage the turbine. The installed configuration of the steam line drain did not match the system description in the Updated Final Safety Analysis Report (UFSAR).

  This issue will remain unresolved pending further review of the Hatch HPCI turbine design and, if appropriate, a complete significance determination by the NRC. (Section 1R21.1 b. (3))

Report Details

1. REACTOR SAFETY

Cornerstones: Initiating Events and Mitigating Systems

1R21 Safety System Design and Performance Capability (71111.21)

.1 System Needs

a. Inspection Scope

General Scope

The team evaluated the plant and operator response to the Unit 2 initiating event “Loss of Station Battery A (Division I) DC power.” This initiating event was selected because it had a relatively high importance in the licensee’s probabilistic risk assessment. The team focused on the installed equipment and operator actions that could initiate the event or would be used to mitigate the event. Another objective was to verify that the licensee’s design and maintenance activities were adequate to minimize the likelihood of this event. A list of specific documents reviewed during this inspection can be found in the attachment to this report.

Process Medium

The team reviewed the licensee’s calculations for water volume in the CST and for net positive suction head for the HPCI pump. This included reviews of system drawings and inspection of installed equipment to compare arrangements and dimensions to those used in the calculations. The team also reviewed the licensee’s calculations supporting the setpoint for the CST level instrumentation which initiates an automatic switchover of the HPCI pump suction from the CST to the suppression pool. This included checking the adequacy of the calculations and comparing calculated values to values in the TS and in the instrument calibration procedures.

The team also performed a failure modes and effects type evaluation of all electrical loads fed from the Unit 2 250/125 volt (V) DC Division I power subsystem. This review specifically included checking which 4160 V main incoming breakers had their DC control power supplied from Division I and which had their control power supplied from Division II. The team also reviewed the Bus 2A, 2B, 2C, and 2D auto-transfer circuit in terms of loss of control power. In addition, the team observed a simulation of a loss of Division I DC power by licensed operators on the plant simulator and compared the response of the simulator to that which would be expected based on a review of design information and plant procedures.

Operator Actions

The team reviewed portions of the emergency operating procedures (EOPs), abnormal operating procedures, annunciator response procedures, and system operating procedures (SOPs) to verify that the operators could perform the necessary actions to respond to a loss of Division I DC power event. The team also observed a simulation of a loss of Division I DC power event by licensed operators on the plant simulator. During the procedure reviews and simulator observations, the team focused on verifying that operators had adequate procedures to: verify or initiate the HPCI system, inhibit the ADS, perform emergency depressurization of the reactor pressure vessel, manually align the RCIC system locally, manually increase control rod drive system flow, and restart the drywell coolers. The team also focused on verifying that operators had adequate instrumentation to perform these actions. In addition, the team checked whether the EOP steps were consistent with those of the Boiling Water Reactor Owners' Group Emergency Procedure Guidelines.

Energy Sources

The team reviewed the electrical design of the HPCI system to verify that a loss of Division I DC power event would not prevent the system from performing its function. Specifically, the team reviewed the HPCI system elementary drawings to determine the effect that loss of the Division I DC power would have on the HPCI system. The team also reviewed appropriate test and design documents to verify that the voltage to HPCI system instrumentation and controls and DC motor operated valves would be adequate and in accordance with equipment specifications and/or test results.

The team also reviewed the maintenance program for the Unit 2 Division II battery, which would supply power in the event of failure of the Division I system (including station Battery A). The inspection considered specific gravity, electrolyte level, temperature, detailed visual inspections, connection resistance, service tests and performance tests. Samples of completed data sheets were reviewed. The team made a detailed visual inspection of the battery room, racks, and cells, and also observed licensee technicians making measurements on the battery. The criteria for the inspection was TS 3.8.4.1 through 3.8.4.6, TS 3.8.6.1 through 3.8.6.3, Institute of Electrical and Electronics Engineers recommended practices, and Electric Power Research Institute guides.

The team reviewed HPCI system drawings and performed a walkdown of the HPCI system to verify correct valve alignment. During the walkdown, the team checked for the inclusion of drains in the steam supply piping to the HPCI turbine to ensure the availability of the steam supply. Additionally, the team reviewed accumulator volume and allowed leakage calculations, system drawings, and design specification sheets to verify the adequacy of the nitrogen supply to operate the ADS safety/relief valves solenoid valves.

Instrumentation and Controls

The team reviewed electrical elementary and logic diagrams depicting the HPCI pump start and stop logic, and permissives and interlocks for valves 2E41-F003, 2E41-F041, 2E41-F042, and 2E41-F004 to ensure that the pump start and shutdown logic, and valve permissives and interlocks, were consistent with the system operational requirements described in the UFSAR.

The team also reviewed the design of protective instrument loops to verify that the instruments would be available in a loss of Division I DC power event. Design drawings and other design documents were reviewed to verify that the as-built instruments' range, accuracy, power supplies, and setpoints were in accordance with design basis documents. The team also reviewed calibration and test records to verify that instrument loops (including trip units) were calibrated in accordance with the calibration program procedures and, if applicable, the TS. The team walked down those instruments that were accessible to verify that they were in good material condition and had been located in accordance with design documents. The instruments reviewed included: 2E51-TE-N066D, Torus high ambient temperature; 2E51-DTIS-N665D, HPCI torus differential temperature high; 2E41-TE-N070B, HPCI equipment high ambient temperature; 2E41-TE-N071B, HPCI pipe room high ambient temperature; 2E41-DPT-N057B, HPCI steam high flow; 2E41-PT-N058B(D), HPCI steam supply low pressure; 2E41-PT-N055B(D), HPCI turbine exhaust diaphragm high pressure; 2B21-LT-N091A thru D, Reactor Pressure Vessel water level 2; 2E11-PT-N094A thru D, drywell high pressure; and 2E41-FIC-R612, HPCI flow indicating controller.

b. Findings

(1) Process Medium

The team identified a green finding and non-cited violation for inadequate design control of the HPCI suction source from the CST. The net positive suction head calculation which determined the CST level setpoint for automatic HPCI system suction switchover from the CST to the suppression pool did not account for vortexing. Vortexing could cause air ingestion into the HPCI system suction and the air could then damage the HPCI pump.

While reviewing Hatch Unit 2 Calculation 495, "HPCI CST Suction Uncovery," Rev. 0, dated April 22, 1985, the team noted that the calculation did not consider the effects of vortexing that could occur in the CST. Calculation 495 supported the TS Table 3.3.5.1-1 allowable value of greater than or equal to 2.61 feet for "Condensate Storage Tank Level - Low." This TS requirement was for emergency core cooling system (ECCS) instrumentation for automatic switchover of the HPCI system suction from the CST to the suppression pool, to ensure an adequate suction source for the HPCI pump. Calculation 495 also supported the corresponding CST level setpoint of 2 feet 10 inches that was specified in Procedure 57SV-SUV-015-2S, HPCI/RCIC Pump Suction Source Instrument Functional Test and Calibration, Rev. 4, Ed. 2.

Calculation 495 and the design drawings of the CST described a 16-inch HPCI system suction pipe (with an inner diameter of 15 5/8 inches) connected to the CST, with the pipe centerline at a CST level of 1.0 foot. The pipe did not protrude into the CST and there was no related anti-vortexing plate in the CST. Calculation 495 described the HPCI suction switchover to involve opening of the suppression pool suction valves (2E41-F041 and F042) followed by closure of the CST suction valve (2E41-F004). The stroke time of each of the valves was stated to be 82 seconds and the total switchover time 164 seconds. Calculation 495 stated that during the 164-second switchover, with a nominal HPCI pump flow of 4250 gallons per minute, the CST level would drop 13 ½ inches. (Note: The CST contained approximately 11,400 gallons per foot of level.)

Calculation 495 also stated that the final CST water level at the end of switchover would be below the top of the HPCI pump suction nozzle on the CST.

Vortexing in pump suction sources is a well known phenomenon. It is discussed in typical textbooks on centrifugal pumps. NRC Regulatory Guide 1.82, “Sumps for Emergency Core Cooling and Containment Spray Systems,” dated June 1974, discussed the need to preventing vortexing. Regulatory Guide 1.82 Rev. 1, dated November 1985, and Rev. 2, dated May 1996, included specific guidance on how to prevent ingestion of air into ECCS suction pipes due to vortexing. That guidance included limiting the Froude number (Fr) to less than 0.8 [where Fr is equal to the inlet pipe velocity (U) in feet per second divided by the square root of {the suction pipe centerline submergence below the water level (S) in feet times gravity (g) in feet per second squared}]. NRC NUREG / CR-2772, “Hydraulic Performance of Pump Suction Inlet for Emergency Core Cooling Systems in Boiling Water Reactors,” dated June 1982, included experiments on suctions from tanks and showed almost no air entrainment with a Fr of 0.8. The BWR Owners’ Group Emergency Procedure Guidelines included guidance on preventing vortexing in ECCS pump suctions from the suppression pool. This guidance included a vortex limit curve based on maintaining Fr less than 0.8. NRC Information Notice (IN) 97-60, “Incorrect Unreviewed Safety Question Determination Related to Emergency Core Cooling System Swapover from the Injection Mode to the Recirculation Mode,” discussed preventing vortexing in the refueling water storage tank to protect the ECCS pumps. By way of industry experience, Dresden Unit 2 Licensee Event Report (LER) 50-237-97-017 addressed preventing vortexing at the HPCI suction nozzle in the CST. This LER stated that, due to a design error, the effect of vortexing on the useable CST volume was not considered in the original design of the HPCI system. It further stated that a Froude number of 0.8 is the industry accepted value to ensure little or no air is entrained in the suction. Dresden subsequently raised the CST low level setpoint for automatic HPCI suction switchover to prevent air entrainment associated with vortexing from degrading the performance or damaging the HPCI pumps.

The NRC team evaluated the Hatch Unit 2 vulnerability to vortexing at the HPCI pump suction from the CST. The team noted that there were no instructions to operators to throttle HPCI pump flow at low CST levels. The team calculated that, with a nominal HPCI pump flowrate of 4250 gpm, a Fr of 0.8 would begin to be exceeded at a CST level of approximately 3 feet 6 inches. The team determined that a Fr of 0.8 would continue to be exceeded after the automatic suction switchover initiated at a CST level of 2 feet 10 inches, until the HPCI suction valve from the CST stroked approximately 2/3 closed and began to throttle flow. The Fr of 0.8 would be exceeded for a total of approximately 1 foot 7 inches of CST level decrease which would take approximately 4 minutes. During that time, a Fr of 1.0 would be exceeded for approximately 1.6 minutes. The team concluded that the generally accepted standards for prevention of vortexing were exceeded by the licensee’s design.

The team analyzed the significance of this issue. The licensee had not included the generally accepted standards for preventing vortexing in pump suction sources in Calculation 495. Also, the licensee had no alternative vortex prevention devices, testing, or analysis to ensure that vortexing would not cause pump damage. As a result, the TS value of 2.61 feet for CST Level - Low instrumentation and the corresponding instrument setpoint of 2 feet 10 inches were not adequate to ensure the prevention of HPCI pump damage by air intrusion from vortexing in the CST. Consequently, the team concluded that failure to account for the effects of vortexing in Calculation 495 constituted a performance deficiency. Since this performance deficiency should have been prevented, was reasonably within the licensee's ability to foresee and correct, and has safety significance, it is considered to be a finding. This finding is associated with the initial design attribute of the mitigating systems cornerstone. This finding affects the mitigating system cornerstone objective because it could affect the capability of the HPCI system to respond to initiating events to prevent undesirable consequences. Therefore, this finding is considered to have greater than minor safety significance. Also, the team considered that this finding has greater significance than similar issues described in NRC Inspection Manual Chapter 0612, "Power Reactor Inspection Reports," Appendix E, Sections 3.a, b, and i. However, licensee use of the non-safety CST as a HPCI pump suction source with the CST at low levels is unlikely because the reactor vessel or suppression pool would generally reach a high level first, where the HPCI pump would be automatically stopped or its suction would be automatically switched to the safety-related suppression pool. In addition, alternate core cooling methods would be available, including RCIC as well as ADS and LPCI. Consequently, this finding is of very low safety significance (Green).

10 CFR 50, Appendix B, Criterion III, Design Control, requires that design control measures shall include provisions to assure that appropriate quality standards are specified and included in design documents. However, the licensee had not included the generally accepted standards for preventing vortexing in pump suction sources in Calculation 495. This violation occurred in 1985 and has existed for 17 years. The violation was identified in 2002 during this inspection. The potential safety consequence is damage to the HPCI pump due to vortexing in the CST. This is being treated as an NCV, consistent with Section VI.A.1 of the enforcement policy and is identified as NCV 50-366/02-06-01; Failure to Consider Vortexing in the Calculation for CST Level for Automatic Switchover of the HPCI Pump Suction. The licensee has entered this issue into their corrective action program as CR 2002010571.

(2) <u>Operator Actions</u>

The team identified a green finding and NCV for an inadequate RCIC SOP that could be used by operators to support the EOPs. The procedure section for local manual start of the RCIC pump with no DC power available, if followed exactly, would have resulted in overspeeding the RCIC turbine with no water flow through the pump and with no cooling water to the RCIC pump.

At the beginning of the inspection, the licensee provided a simulator demonstration of how the plant and operators would respond to a loss of Division I DC power if the HPCI pump was out of service for maintenance or failed to operate. During the demonstration, the operators simulated local manual alignment of the RCIC system for injecting CST water into the reactor vessel. The local manual operation was used because the RCIC system normally uses Division I DC power for pump speed control and for operation of motor operated valves.

The team reviewed Section 7.3.8, “Local Manual RCIC Operation,” of Procedure 34SO-E51-001-2S, “Reactor Core Isolation Cooling (RCIC) System,” Revision 21.3. The team reviewed this procedure with senior reactor operators (SROs) and conducted a walkdown with a reactor operator. Two SROs showed the team that the EOPs referenced the RCIC SOP for use to supply water to the reactor vessel. They stated that, during the loss of Division I DC power event, operators would be able to use the local manual operation section of the RCIC procedure. During the procedure review and walkdown, the team noted that the procedure steps were written to first align all necessary valves and then to slowly open Valve 2E51-F524, RCIC Pump Trip/Throttle Valve, to start the RCIC turbine. The last step assumed the trip/throttle valve was closed, however, the team found that the trip/throttle valve was normally aligned in the open position. Had the operator performed the steps as written, the RCIC turbine would have been supplied with steam five steps earlier, when Valve 2E51-F045, RCIC Steam Inlet, was opened. The team concluded that this action would result in overspeed and trip of the RCIC turbine. The team noted that, per the procedure, when the RCIC steam inlet valve was opened, the RCIC pump discharge valve and cooling water valves had not yet been opened and the RCIC pump minimum flow valve was closed. The team concluded that the RCIC turbine would overspeed with no water flow through the pump and with no cooling water.

During the walkdown with a reactor operator, the team found that the operator had not performed the local manual operation section of the RCIC SOP before. The team later learned that operators had not received training on that section of the procedure. The team further noted that performance of the section would be difficult for one operator because, while most of the valves were located in the RCIC system diagonal, two valves were in the torus room and one valve was located in the HPCI system room. In addition, with a loss of Division I DC power and the resulting loss of non-vital AC power, lighting would be such that one operator would need to hold a flashlight while another operated the valves. Furthermore, one operator could not operate the trip/throttle valve and monitor the handheld tachometer at the same time because of the distance between the two points. The team concluded that the lack of operator training and the difficulties that would be encountered by one operator would unnecessarily lengthen the amount of time needed to put the RCIC system in service during an emergency condition.

The team also noted that the steps in the procedure section for local manual operation of RCIC were not organized in an efficient manner. Because the valves operated by the procedure were motor operated valves intended to be operated manually, i.e. without electric power, the procedure included steps to open the power supply breakers before manipulating the valves. However, the steps were ordered in such a way that the operator was required to climb the stairs in the RCIC system diagonal each way at least twice to complete the operations. The team concluded the inefficient organization of the procedure would unnecessarily lengthen the amount of time needed to put the RCIC system in service during an emergency condition.

The team also noted a procedural weakness with steps designed to manually align the RCIC system for pressure control only. These steps opened the test return line to the CST and intended to close the injection path to the reactor vessel. However, because the procedure assumed that the trip/throttle valve was closed when it was not, had the steps for pressure control been performed as written an injection path to the reactor vessel would be open at the same time as the path to the CST. The team concluded this would only be an inconvenience unless check valves in the injection line to the reactor vessel leaked, which would inappropriately allow water from the reactor vessel to flow into the CST.

The team reviewed the procedure development package for Section 7.3.8 of Procedure 34SO-E51-001-2S and learned that the section was implemented for physical security reasons or for conditions where AC and DC power were both lost. Reviews had been conducted by operations, compliance, and ALARA departments, and a procedure validation had not been performed. The operations department review had not addressed any of the potential problems mentioned above. The licensee did not conduct a technical review of Section 7.3.8 of Procedure 34SO-E51-001-2S prior to issuance.

The team analyzed the significance of this issue. Because the RCIC System was credited as mitigating equipment for several initiating events in the NRC Risk Informed Notebook for the Hatch Plant, this issue affected the mitigating systems cornerstone. The capability of the RCIC system to mitigate these events would be affected, due to procedure quality problems, if the local manual operation section of Procedure 34SO-E51-001-2S were implemented. This makes this issue more than minor. However, because the likelihood of losing Division I DC power was low and because other core cooling methods could be available, including HPCI as well as ADS and LPCI, this issue was of very low safety significance (Green).

TS 5.4.1, Procedures, requires that written procedures be established, implemented, and maintained covering activities outlined in Appendix A of Regulatory Guide 1.33, Revision 2. Regulatory Guide 1.33, Appendix A, Section 4.g, requires a procedure for the Reactor Core Isolation Cooling System. From January 5, 1995, to September 13, 2002, Procedure 34SO-E51-001-2S, Reactor Core Isolation Cooling System, was inadequate in that the section for local manual operation would have resulted in overspeeding the RCIC pump with no water flow through the pump and with no cooling water to the pump. This is being treated as an NCV, consistent with Section VI.A.1 of the Enforcement Policy and is identified as NCV 50-366/02-06-02: Inadequate RCIC System Operating Procedure. This issue is in the licensee's corrective action program as part of CR 2002009254 and CR 2002009261.

(3) Energy Sources

An unresolved item (URI) was opened for further NRC review of the Hatch HPCI turbine design and, if appropriate, a complete significance determination of conditions that could result in the loss of the HPCI pump during a certain plant events. These conditions could occur because of improper design control of the HPCI System.

The HPCI system is a Division II system. In reviewing the HPCI System Elementary Drawings, H-27664 through H-27672, the team observed three circuits with power supplied by Division I DC power. The team examined each of these to determine the effect that loss of Division I DC power would have on the HPCI System. One circuit powered the logic for HPCI system isolation and another circuit powered the logic for HPCI system initiation on low reactor water level. The team determined that loss of

Division I DC power would not affect either of these functions because a second circuit, powered from the Division II Battery, would be available.

The third circuit provided power for Valves 2E41-F028, Steam Line Drain Isolation Valve, and 2E41-F025, Condensate Pump Isolation Valve. The team determined that loss of power to Valve 2E41-F025 would have no effect on the HPCI System because the valve isolated a drain from the condensate pump and failed closed on loss of power. However, the team determined that loss of power to Valve 2E41-F028 would affect the HPCI system. Valve 2E41-F028 was an isolation valve on the drain line from the HPCI Steam Line Drain Pot. This air operated valve was normally kept open to allow condensate collected in the pot to drain through an orifice to the main condenser. This was done to prevent water buildup in the HPCI turbine steam supply line. If the Division I Battery were lost, this valve would close and isolate the drain. This would result in water buildup in the drain pot and eventually in the steam line itself. The drain pot contained a level switch that, upon high level, opened a bypass around the orifice in the drain line, however, flow through this bypass line was also required to go through Valve 2E41-F028. The level switch also provided an alarm in the control room.

UFSAR Sections 7.3.1.2 and 7.3.1.2.1.6 provided separation criteria for the HPCI system instrumentation and control electrical power supplies. These sections stated that power was provided by Division II except for Valve 2E41-F002, Inboard Isolation Valve. Because the power supply for the logic circuit to Valve 2E41-F028 was from Division I, the team concluded that the design did not meet the separation criteria specified in the UFSAR. Additionally, UFSAR Sections 6.3.2.2.1 and 7.3.1.2.1.3 stated that the steam line drain pot and isolation valves were intended to prevent water buildup in the HPCI steam supply line. The team further concluded that, under certain conditions, the design would not prevent water from collecting in the steam line as intended by the UFSAR and that this could affect HPCI system reliability.

The team also noted that even if power from Division I DC power were not lost, the logic design of Valve 2E41-F028 could affect the HPCI System in certain circumstances. The valve was interlocked with Valve 2E41-F001, HPCI Turbine Steam Supply Valve, such that Valve 2E41-F028 closed when Valve 2E41-F001 was not fully closed (ie. when Valve 2E41-F001 opened to start the HPCI pump). This was intended to isolate the steam line drain pot when the HPCI turbine was running. However, under certain conditions, such as high reactor water level, the HPCI turbine would trip without closing Valve 2E41-F001. In these conditions the HPCI turbine would be shut down with the drain pot isolated. This would result in condensate buildup in the steam line. Also in these conditions, the drain pot level switch would not provide an alarm on high level because the alarm was also interlocked with Valve 2E41-F001.

The licensee addressed the significance of the issue and concluded that the HPCI turbine would not be damaged by as much as 600 gallons of water entering and leaving the turbine. They referenced an article from the October, 1973, issue of “Power Magazine” in which the turbine manufacturer conducted a series of tests on a similar turbine. The tests included a simulated quick start with the steam line full of water along with a series of water slugs injected into the turbine while it was operating. No damage resulted from the tests. The team reviewed the article and noted that the tests were conducted using subcooled water and saturated steam at 600 psig because damage was more likely with subcooled water than with saturated water or a two phase mixture. Because the HPCI System was designed to operate at pressures exceeding 1000 psig, the team questioned the validity of the licensee's conclusion.

The licensee also addressed a concern raised by the team regarding water passing through the HPCI turbine that was described in LER 50-254/93-005. In this event, water entered the HPCI turbine at Quad Cities Unit 1 and resulted in a failure of the HPCI turbine exhaust rupture disc and loss of secondary containment. The licensee referenced two letters from a HPCI turbine expert regarding this failure and other similar failures on the RCIC turbine at the LaSalle Station. The licensee stated that the previous rupture disc failures were due to direct impingement of water on the disc and attributed it to the exhaust piping arrangement that placed the rupture discs in a direct path from the turbine outlet. The licensee further explained that the HPCI turbine at Plant Hatch was not susceptible to this failure because the piping arrangement was not the same. The team reviewed the letters and LER and confirmed that the rupture disc piping arrangement at Plant Hatch was different from that at Quad Cities. However, the team also noted that Quad Cities Unit 1 was shutdown due to possible damage to the HPCI turbine and that one of the letters stated that the thrust bearing on the Quad Cities HPCI turbine was damaged. The licensee stated that the Hatch HPCI turbine design was not the same as that of Quad Cities and that the thrust bearing would not be damaged by water entering the turbine. However, the team was not able to verify this assertion before issuing this report.

This issue will remain unresolved pending further review of the Hatch HPCI turbine design and a complete significance determination by the NRC. This issue is identified as URI 50-366/02-06-03: Design Control of HPCI System Steam Line Drain Pot.

(4) <u>Instrumentation and Controls</u>

No findings of significance were identified.

.2 <u>System Condition and Capability</u>

a. <u>Inspection Scope</u>

<u>Installed Configuration</u>

The team visually inspected compartments 02B, 03B, 08A, and 08B on 250 V DC Motor Control Center (MCC) 2B to verify that the installed components (i.e., breakers, fuses, and overload heaters) were the correct type and size as specified by the design drawings. These MCC compartments were associated with HPCI DC motor operated valves 2E41-F003, -F004, -F041, and -F042. The team also looked to see if the breakers had the proper trip settings, whether the compartments were free of dirt and debris, and whether wiring, contactors, and terminations were in good physical condition with no signs of deterioration.

The team also made a detailed visual inspection of the Division II Battery since this battery would supply power in the event of failure of the Division I Battery. The team checked the following attributes and potential problems: float voltage, copper contamination, cracked jars, leaking or deformed jar seals, plate color and alignment, separator intact and extends beyond plate, electrolyte level and contamination, sediment, nameplate, stamped codes indicating date and location of manufacturing, evidence of mossing at the plate straps, evidence of leaks, room temperature, heating and ventilating equipment, and corrosion on connectors.

Operation

The team walked down portions of the HPCI system to verify that it was aligned so that it would be available for operators to mitigate a loss of Division I DC power event. During this walkdown, the team compared valve and breaker positions with those specified in the system operating procedure lineups, and observed materiel condition of the plant to determine if it would be adequate to support operator actions to mitigate a loss of Division I DC power event. The team also compared the actual configuration of the Division I and II 125/250 VDC systems with the design drawings, including reading of all instruments such as voltmeters, ammeters, and ground detection equipment.

Design

The team reviewed the coordination of overcurrent protection devices on the Division I system from the viewpoint that relatively minor faults could cascade to event initiators. Modifications and equipment replacements were considered in relation to possible alteration of time-current characteristics of the protection devices. The team examined installed overcurrent protection devices and compared set points and style numbers to design document information. The team also reviewed the battery loading and sizing calculation for the Division II battery. In addition, the team reviewed the design of the ground detection equipment as compared to system requirements for sensitivity.

The team also reviewed the design of the control circuit for the safety/relief valves to check if it implemented the system operation described in the design basis documents. Circuit response to loss of DC power was also considered. The team reviewed the licensee’s system voltage analysis in relation to ensuring adequate voltage for the safety/relief valve solenoid operated air valves.

Also, the team reviewed a sample of completed plant design change packages (i.e., DCRs 00-0002, 00-0003, 90-0109, and 92-0091) to verify that the changes did not degrade the design or functional capability of the system as described in the UFSAR and design basis documents.

Testing

The team reviewed the licensee’s program and methodology for periodically testing the safety/relief valve control circuits. The team also reviewed the licensee’s program and methodology for periodically testing a check valve in the HPCI injection path in terms of the design basis function of that check valve.

The team reviewed calibration procedure 57SV-SUV-015-2S, “HPCI/RCIC Pump Suction Source Instrument Functional Test and Calibration,” Rev. 4ED2 and completed calibration test results on the condensate storage tank level switch channels (i.e., 2E41-

LS-N002 and N003) to verify that the instrument setpoints were in accordance with the instrument setpoint index and the TS. The team reviewed “as-found” and “as-left” test data to verify that out of tolerance conditions were being properly evaluated for system impact by the licensee and that all “as-left” data met the procedure acceptance criteria. The dates of calibration were reviewed to verify that the calibration intervals were in accordance with technical specifications. The team also reviewed the calibration test procedures to verify that the interlocks for HPCI valves 2E41-F041, F042, and F004 were properly verified during testing. The other calibration procedures and test records reviewed are listed in the List of Documents Reviewed section of this report.

b. Findings

No findings of significance were identified.

.3 Selected Components

a. Inspection Scope

Component Degradation

The team reviewed system health reports for the DC systems to ascertain whether significant problems had been occurring. In addition, the team reviewed the summary of problem reports related to the DC system. This report covered the period from 1996 to a few weeks before the beginning of the inspection. From the summary a sample of actual problem reports were requested and reviewed. As deemed necessary, additional information related to the extent of condition or corrective actions for the identified problems was requested and evaluated.

The team reviewed the maintenance history over the past five years on a sample of risk significant HPCI motor operated valves to assess the licensee’s actions to verify and maintain the safety function, reliability, and availability of the components in the system. The team also reviewed the torque switch setting guide sheets for several motor operated valves to verify that the voltages used in analyzing the adequacy of torque switch settings were consistent with those described in design basis calculation SENH-02-005, Station Service Battery 2B (2R42-S001B) - DC MOV Voltage Analysis, Revision 0. The team reviewed the minimum and maximum voltage requirements for various other components (e.g., relays and power supplies) fed from associated HPCI panels 2H11-P620, 2H11-P928, and 2H11-P926 to verify that the load voltages were consistent with manufacturers or special testing requirements.

Equipment/Environmental Qualification

The team reviewed the qualification test report for the Division II battery.

Equipment Protection

The team evaluated the potential effects of external events on the DC system.

Component Inputs/Outputs

The team reviewed the acceptable range of input voltage for the battery chargers as compared to the calculated voltage regulation and expected transients on the 600 V system.

Operating Experience

The team reviewed the licensee's evaluations for the operating experience reports listed in the List of Documents Reviewed section of this report to verify that the licensee's review and actions were appropriate. The team also reviewed the reactor trip and transient of January 26, 2000, as described in NRC Information Notice 2000-01 to verify that lessons learned for the HPCI and RCIC Systems were properly implemented.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification And Resolution Of Problems

a. Inspection Scope

The team reviewed licensee audits and assessments related to the inspection scope and also reviewed selected samples of condition reports related to equipment that could initiate or would be used to mitigate a loss of Division I DC power event, to assess the adequacy and timeliness of the licensee's corrective actions. The team also reviewed corrective maintenance work orders on risk significant equipment to evaluate failure trends.

b. Findings

This inspection identified problems with Section 7.3.8, "Local Manual RCIC Operation," of Procedure 34SO-E51-001-2S, "Reactor Core Isolation Cooling (RCIC) System," Revision 21.3. See Section 1R21.1.b (2) for details. Recovery from a reactor scram and transient on January 26, 2000 was complicated by similar problems in a different section of this procedure. The January 26 event is described in NRC Information Notice 2000-01, "Operational Issues Identified in Boiling Water Reactor Trip and Transient." Subsequent NRC inspections are documented in NRC Inspection Report 50-321,366/00-01, and NRC Inspection Report 50-321,366/00-02. As a result of the problems with the RCIC procedure during that event, the NRC issued NCV 50-321,366/00-02-02, Inadequate Procedure for Restarting RCIC Following a High Reactor Water Level Trip. The team noted that the licensee's corrective action from the 2000 event missed an opportunity to identify and correct deficiencies, with the SOP section for local manual RCIC operation, that the NRC identified during this inspection.

4OA6 Meetings

Exit Meeting Summary

On September 13, 2002, the team presented the inspection results to Mr. P. Wells and other members of licensee management and staff at the conclusion of the onsite inspection. On October 18, 2002, the team presented revisions to the findings by telephone to Mr. S. Tipps and other members of the licensee's staff. The licensee acknowledged the findings presented.

The licensee's representatives were aware that some proprietary information had been reviewed by the team, however, no proprietary information is contained in this report.

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

Licensee

J. Betsill, Assistant General Manager - Support
K. Breitenbach, Engineering Supervisor
E. Burkett, Operations Support Superintendent
R. Dedrickson, Operations Manager
J. Lewis, Training and Emergency Preparedness Manager
T. Metzler, Nuclear Safety and Compliance Engineering Supervisor
J. Robertson, Engineering Group Supervisor and Acting Manager
V. Shaw, Project Engineer
W. Snider, Electrical Supervisor
S. Tipps, Nuclear Safety and Compliance Manager
C. Tully, Licensing Engineer
P. Wells, Plant General Manager

NRC

J. Munday, Senior Resident Inspector
C. Ogle, Chief, Engineering Branch I

ITEMS OPENED, CLOSED, OR DISCUSSED

Opened

URI 50-366/02-06-03	Design Control on HPCI System Steam Line Drain Pot (Section 1R21.1.b (3))

Opened and Closed

NCV 50-366/02-06-01	Failure to Consider Vortexing in the Calculation for CST Level for Automatic Switchover of the HPCI Pump Suction (Section 1R21.1.b (1))
NCV 50-366/02-06-02	Inadequate RCIC System Operating Procedure (Section 1R21.1.b (2))

ACRONYMS USED

ADS	automatic depressurization system
CST	condensate storage tank
DC	direct current
ECCS	emergency core cooling system
EOP	emergency operating procedure
Fr	Froude number
HPCI	high pressure coolant injection
LER	Licensee Event Report
LPCI	low pressure coolant injection
MCC	motor control center
RCIC	reactor core isolation cooling
SOP	system operating procedure
SRO	Senior Reactor Operator
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	unresolved item
V	volt

LIST OF DOCUMENTS REVIEWED

Condition Reports Resulting from this Inspection

CR 2002008662, Labeling for the DC Control Power Potential Lights was Not Easy to Understand
CR 2002008673, Five Flourescent Lights in the 2A Station Battery Room are Not Working Properly
CR 2002008690, Minor Discrepancies in the Data Packs for the Station Service Battery Chargers
CR 2002008733, Oil Puddle Under Unit 2 HPCI Pump
CR 2002008737, Battery Rack Brace is Mis-aligned
CR 2002008739, Minor Error on Drawing H-24748
CR 2002008766, Tape Found in Electrical Cabinets
CR 2002008767, Difficulty Closing Electrical Cabinet Doors
CR 2002008795, Error in Lesson Plan on DC Electrical Distribution
CR 2002008818, Design Calculations Indicate that In Certain Conditions DC Voltage to SRVs Could be Less Than The Voltage For Which SRVs are Qualified
CR 2002009052, Minor Error in Calculation SENH 94-21 Regarding Battery Resistance
CR 2002009146, Calculation SENH 94-002 Incorrectly Indicates that Topaz Inverters are Currently in Use
CR 2002009180, Locking Nut on Breaker MCC Cubicle is Missing
CR 2002009214, Calculation SENH 91-011 Contained Incorrect Trip Ratings for Two Breakers
CR 2002009254, Procedural Steps in section 7.3.8 of 34SO-E51-001-1/2 Could Result in a Water Hammer if Performed as Written
CR 2002009261, Procedural Problems with 34SO-E51-001-2S, Section 7.3.8, Local Manual Operation of RCIC
CR 2002009263, Non-safety HPCI Turbine Oil Level Switch is not Electrically Isolated from Safety-related HPCI Control Logic
CR 2002010571, Potential for Vortexing in HPCI Suction from the CST

Procedures

34AB-R42-001-0S, Location of Grounds, Rev. 1.2
34SV-B21-006-2S, Feedwater Check Valve Operability, Rev. 1.6
42SV-R42-003-0S, Battery Inspection, Rev. 8.1
42SV-B21-003-2S, Automatic Depressurization System Logic System Functional Test, Rev.9
52SV-R42-001-2S, Battery Pilot Cell Surveillance, Rev. 19.5
52SV-R42-002-2S, Battery/Individual Cell Surveillance, Rev. 15.1
52GM-B21-005-0, Main Steam Relief Valve Maintenance, Rev. 18.0
10AC-MGR-003-0, Procedure Processing, Rev. 20.0
10AC-MGR-027-0, Applicability Determination, Rev. 0.1
30AC-OPS-006-0, Verification Program for Emergency Operating Procedures, Rev. 5.1
30AC-OPS-007-0, Emergency Operating Procedures Revision Requirements, Rev. 7.1
30AC-OPS-013-0, Use of Emergency Operating Procedures, Rev. 9.1
31EO-EOP-010-2, RC, RPV Control (Non-ATWS), Rev. 7
31EO-EOP-015-2, CP-1, Alternate Level Control, Steam Cooling, & Emergency RPV Depressurization, Rev. 6

31EO-EOP-012-2, PC-1, Primary Containment Control, Rev. 4
31EO-EOP-107-2, Alternate RPV Pressure Control, Rev. 4.2
31EO-EOP-108-2, Alternate Depressurization, Rev. 5.2
31EO-EOP-114-2, Preventing Injection into the RPV From Core Spray and LPCI, Rev. 0.1
31EO-EOP-100-2, Miscellaneous Emergency Overrides, Rev. 6.2
31EO-EOP-110-2, Alternate RPV Water Level Control, Rev. 2.2
34AB-B21-002-2, RPV Water Level Corrections, Rev. 6.6
34AB-C71-001-2, Scram Procedure, Rev. 9.6
34AB-C32-001-2, Reactor Water Level Above +60 Inches, Rev. 0.1
34AB-R22-001-2, Loss of DC Busses, Rev. 2.3
34AC-OPS-011-2, Validation Program for Emergency Operating Procedures, Rev. 1.1
34SO-E41-001-2, High Pressure Coolant Injection System, Rev. 21.2
34SO-E51-001-2, Reactor Core Isolation Cooling System Operating Procedure, Rev. 21.3
34SO-E51-001-2, Reactor Core Isolation Cooling System Operating Procedure, Rev. 17
34SO-C11-005-2, Control Rod Drive Hydraulic System, Rev. 24.0
34AR-601-103-2, Annunciator Response Procedure, HPCI Turbine Trip, Rev. 2.1
34AR-601-110-2, Annunciator Response Procedure, HPCI Turbine Inlet Drain Pot Level High, Rev. 4.1
34AR-601-126-2, Annunciator Response Procedure, HPCI Logic Bus Power Failure, Rev. 1
57SV-CAL-003-2, ATTS Transmitter Channel Calibration, Rev. 16.4

Drawings

A-26497, Unit 2 Instrument Setpoint Index, Rev. 66
A-43830, Sheets: 2E41-F004, 2E41-F041, 2E41-F003, and 2E41-F042, dated 3/6/02
A-4483701, Magnetic Only Molded Case Circuit Breaker Instantaneous Trip Setting Guidelines, Rev. 0
A-51687, Sheet 1, Motor Control Centers & Local Starters Thermal Overload Relay Contact Bypass Jumper Wiring, Rev. 4
A-51371, Sheet 82, Unit 2 Master Fuse List (Non Control Room Fuses), Rev. 043
B-27050, pages 1 thru 3, Hatch Unit 2 Motor Control Center (MCC) Equipment List Reactor Building 250 Volt DC MCC 2B 2R24S022, Rev. 14
H-24742, HPCI System Logic Diagram, Sheet 1, Rev. 3
H-24743, HPCI System Logic Diagram, Sheet 2, Rev. 3
H-24744, HPCI System Logic Diagram, Sheet 3, Rev. 3
H-24745, HPCI System Logic Diagram, Sheet 4, Rev. 2
H-24746, HPCI System Logic Diagram, Sheet 5, Rev. 4
H-24747, HPCI System Logic Diagram, Sheet 6, Rev. 3
H-24748, HPCI System Logic Diagram, Sheet 7, Rev. 5
H-24749, HPCI System Logic Diagram, Sheet 8, Rev. 5
H-27664, HPCI System 2E41 Elementary Diagram Sheet 1, Rev. 39
H-27665, HPCI System 2E41 Elementary Diagram Sheet 2, Rev. 35
H-27666, HPCI System 2E41 Elementary Diagram Sheet 3, Rev. 23
H-27667, HPCI System 2E41 Elementary Diagram Sheet 4, Rev. 29
H-27668, HPCI System 2E41 Elementary Diagram Sheet 5, Rev. 24
H-27669, HPCI System 2E41 Elementary Diagram Sheet 6, Rev. 29
H-27670, HPCI System 2E41 Elementary Diagram Sheet 7, Rev. 25

H-27671, HPCI System 2E41 Elementary Diagram Sheet 8, Rev. 26
H-27672, HPCI System 2E41 Elementary Diagram Sheet 9, Rev. 26
H-24100, HPCI System 2E41 Elementary Diagram Sheet 10, Rev. 17
H-51689, HPCI System 2E41 Elementary Diagram Sheet 11, Rev. 2
H-24401, Elementary Diagram ATTS System 2A70, Sheet 1, Rev. 5
H-24422, Elementary Diagram ATTS System 2A70, Sheet 22, Rev. 4
H-24423, Elementary Diagram ATTS System 2A70, Sheet 23, Rev. 5
H-24425, Elementary Diagram ATTS System 2A70, Sheet 25, Rev. 4
H-24426, Elementary Diagram ATTS System 2A70, Sheet 26, Rev. 5
H-24427, Elementary Diagram ATTS System 2A70, Sheet 27, Rev. 3
H-24429, Elementary Diagram ATTS System 2A70, Sheet 29, Rev. 5
H-24430, Elementary Diagram ATTS System 2A70, Sheet 30, Rev. 3
H-24431, Elementary Diagram ATTS System 2A70, Sheet 31, Rev. 3
H-24432, Elementary Diagram ATTS System 2A70, Sheet 32, Rev. 6
H-27877, HPCI System 2E41 Motor Operated Valves & Misc. Devices External Connection Diagram, Rev. 33
H-21084, Condensate Make-up and Storage at Tank Outside Building, Rev. 14
H-23406, Sheet 1, Inverter and Charger Alarms Elementary Diagram [ground detection], Rev. 27
H-23390, Sheet 1, Single Line Diagram 125/250V DC Station Service - Division I Unit 2, Rev. 46
H-23390, Sheet 2, Single Line Diagram 125/250V DC Station Service - Division I Unit 2, Rev. 15
H-23624, Sheet 1, Normal Station Service 4160V Supply Elementary Diagram System, Rev. 15
H-23624, Sheet 2, Normal Station Service 4160V Supply Elementary Diagram System, Rev. 18
H-23624, Sheet 3, Normal Station Service 4160V Supply Elementary Diagram System, Rev. 15
H-23624, Sheet 4, Normal Station Service 4160V Supply Elementary Diagram System, Rev. 20
H-23350, Unit 2 Master Single Line Diagram, Rev. 8
H-26839, HPCI System 16'' Suction From Torus and Condensate Tank, Rev. 1
H-26130, HPCI System Sections, Rev. 7
H-26066, Sheet 1, Drywell Pneumatic System P&ID, Rev. 26
H-27470, Automatic Depressurization System 2B21C Elementary Diagram Sheet 1 of 6, Rev. 21
H-27471, Automatic Depressurization System 2B21C Elementary Diagram Sheet 2 of 6, Rev. 19
H-27472, Automatic Depressurization System 2B21C Elementary Diagram Sheet 3 of 6, Rev. 28
H-27473, Automatic Depressurization System 2B21C Elementary Diagram Sheet 4 of 6, Rev. 21
H-27404, Automatic Depressurization System 2B21C Elementary Diagram Sheet 5 of 6, Rev. 5
H-27403, Automatic Depressurization System 2B21C Elementary Diagram Sheet 6 of 6, Rev. 3
H-28023, Sheet 2, Drywell Pneumatic System P&ID, Rev. 7
H-26071, Safeguard Equip. Emergency Cooling System P&ID, Rev. 9
H-26051, Sheet 2, Reactor Building - Plant Service Water System, Rev. 38
H-26000, Nuclear Boiler System P&ID, Sheet 1, Rev. 38
H-26001, Nuclear Boiler System P&ID, Sheet 2, Rev. 35
H-26189, Nuclear Boiler System P&ID, Sheet 3, Rev. 18

H-26006, Control Rod Drive System P&ID, Sheet 1, Rev. 22
H-26007, Control Rod Drive System P&ID, Sheet 2, Rev. 34
H-26020, HPCI System P&ID, Sheet 1, Rev. 42
H-26021, HPCI System P&ID, Sheet 2, Rev. 30
H-26023, RCIC System P&ID, Sheet 1, Rev. 30
H-26024, RCIC System P&ID, Sheet 2, Rev. 25
H-26084, Primary Containment Purge & Inerting System P&ID, Rev. 30
H-26284, Class IE Analog Signal Conversions/Isolation System IED, Rev. 6
H-26991, Feedwater Control System Turbine Driven Feedpumps IED, Rev. 11
H-27523, Feedwater Control System 2C32 Elementary Diagram, Sheet 5 of 6, Rev. 17
H-27233, Drywell Pneumatic System 2P70 Elementary Diagram, Sheet 3 of 3, Rev. 5
H-27770, Drywell Cooling System 2T47 Elementary Diagram, Sheet 1 of 4, Rev. 22
H-27796, Drywell Chilled Water System 2P64 Elementary Diagram, Sheet 1 of 4, Rev. 26
H-27797, Drywell Chilled Water System 2P64 Elementary Diagram, Sheet 2 of 4, Rev. 24
H-27675, RCIC System 2E51 Elementary Diagram, Sheet 3 of 9, Rev. 25
S-26837, Isometric HPCI, Rev. 24
SX-21428, Condensate Storage Tank, Rev. 7
SX-24226, High Pressure Coolant Injection, Rev. 6

Completed Performance/Surveillances Test Procedures

52SV-R42-002-2S, Individual Cell Surveillance [2R42S001A], Rev. 15.1, dated 4/5/02
52SV-R42-002-2S, Individual Cell Surveillance [2R42S001B], Rev. 15.1, dated 6/28/02
42SV-R42-007-0S, Battery Charger Capacity Test [on 2R42S029], dated 7/15/98, 2/20/00 and 8/19/01
42SV-R42-007-0S, Battery Charger Capacity Test [on 2R42S030], dated 7/14/98, 2/20/00 and 8/19/01
42SV-R42-007-0S, Battery Charger Capacity Test [on 2R42S031], dated 7/15/98, 2/20/00 and 8/19/01
42SV-R42-009-0S, Combined Service-Performance Test and Modified Performance Test [on 2R42-S001B], dated 3/19/00 and 9/21/01

Completed Functional Tests and Calibrations

42SV-E41-002-2S, HPCI Logic System Functional Test, performed 9/17/01
57SV-CAL-004-2S, ATTS RTD Channel Calibration, Rev. 4 performed on 8/16/2000 and 10/19/98
57SV-SUV-014-2S, ATTS Panel 2H11-P928 Channel FT&C, Rev. 14ED5 performed on 6/26/01 and 3/24/01
57SV-SUV-004-2, ATTS RTD Channel Calibration, Rev. 4 performed 8/10/00 and 10/19/98
57SV-SUV-015-2S, HPCI/RCIC Suction Source Instrument FT&C, Rev. 4ED2 performed 1/8/01, 4/4/01 and 7/1/01
57SV-CAL-003-2S, ATTS Transmitters with 24 Month Calibration Frequency, Rev. 5 performed 1/9/01 and 6/4/99
57SV-CAL-003-2S, ATTS Transmitter Channel Calibration Data Sheet, Rev. 16 performed 1/9/01, 4/12/01, 2/11/00, and 6/3/99
57IT-CAL-001-2S, HPCI Turbine Control FT&C, Rev. 7 performed on 5/17/01 and 1/12/00

57CP-CAL-226-2S, Yokogawa Controller Calibration, Rev. 4 completed 3/24/97

Completed Preventive Maintenance Procedures

52GM-B21-005-0S, Main Steam Relief Valve Maintenance for 2B21-F013A, C, D, H, and K, Rev. 16, dated 10/05/01

Calculations

52-2E52-26, Setpoint Calculation for 2E51-DTIS-N665A, B, C, D, Rev. 2
52-2E51-26, Setpoint Calculation for 2E51-DTIS-N665A - D, Rev. 2
SINH 90-017, Calculation of Setpoints to be used for ATTS loops using Rosemount 1154 transmitters with 24 month calibration frequency, Rev. 5
SENH-94002, Evaluate Topaz/Abacus Inverters With Respect To Degraded DC Voltages, Rev.0
SENH 02-005, Station Service Battery 2B (2R42-S001B) - DC MOV Voltage Analysis, Rev. 0
SENH 94-008, Safety Related MOV TOL Heater Element Sizes, Rev. 1
H-85-13, ATTS Setpoint Calculation for Rosemount 1154 transmitters with a 24 month calibration interval for drywell pressure high instruments, Rev. 0
Bechtel Calculation E0087, Rev. 0, Attachment E, sheet E15 (9/11/02)
SENH 93-025, Station Service Battery 2B (2R42-S001B) Sizing and Voltage Profile, Rev. 5, dated 8/23/02
6511-99, Voltage Drops in DC Control Circuits, Rev. 1, dated 1/15/90
SENH 02-030, Minimum Design Basis Operating Voltage for ADS and LLS SRV Solenoids Unit 1, Rev. 0, dated 9/7/02
SENH 02-031, Minimum Design Basis Operating Voltage for ADS and LLS SRV Solenoids Unit 2, Rev. 0, dated 9/7/02
SENH 91-011, Coordination Study for 250 VDC Switchgear 2R22S016 & 2R22S017, Rev. 2, dated 3/1/99
SENH 92-101, Adequacy of Cable Size for Station Service Battery Chargers 2R24-S026, S027, S028, Rev. 0, dated 3/16/92
Calculation 495, HPCI CST Suction Uncovery, Rev. 0
Calculation 352, HPCI Room Heat Load - Unit #2, Rev. 0
Calculation 332, ADS Check Valve Leakage Criteria, Rev. 0
Calculation 384, ADS Valve Accumulator Leakage, Rev. 0
Calculation 410, SRV Mass Flow Rate vs Reactor Pressure, Rev. 0
Calculation 96, NPSH Limits - HPCI & RCIC Pumps, Rev. 7
Calculation 52-2E41-1, LSN002 and LSN003 Condensate Storage Tank Level Low, Rev. 0

Maintenance Work Orders

MWO 2010252401, Feedwater Check Valve 2B21-F076A Will Not Close, dated 09/16/01
MWO 29200389, HPCI Pump Impeller Change, dated 01/25/92
MWO 20103440, Repair HPCI System Oil Leaks, scheduled 09/17/02
MWO 20002245, Valve 2E41-F3040 Repair Due to Unavailabity to Regulate Pressure, dated 05/18/01

MWO 20001751, Adjustment of Pressure Control Valve 2E41-F3040 While HPCI is Running, dated 01/09/01
MWO 20000472, Calibrate HPCI Gauges 2E41R768 and 2E41R769, 04/20/00
MWO 2970356201, Changed Breaker Trip Setting in Frame 02A of MCC 2R24-S022, dated 4/22/98
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MWO 2010158001, Investigate Ground Fault, dated 1/7/02
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MWO 2970356701, Changed Breaker Trip Setting in Frame 08A of MCC 2R24-S022, dated 4/21/98
MWO 2960272901, Replace Valve Operator Motor and Breaker, dated 2/27/97
MWO 2960188602, Perform 18 Month Grease Inspection, Replace Torque Switch, Spring Pack and Test, dated 2/27/97
MWO 2970356801, Change Breaker Trip Setting in Frame 08B of MCC 2R24-S022, dated 4/21/98
MWO 2960273001, Replace Existing Valve Operator Motor and Breaker Per WPS 96-006-E004, dated 3/21/97
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<u>UFSAR</u>

UFSAR Section 6.3.2.2.1, HPCI System, Rev. 20
UFSAR Section 6.3.2.2.2, Automatic Depressurization System (ADS), Rev. 20
UFSAR Section 7.3.1.2, ECCS System Descriptions, Rev. 19
UFSAR Table 5.2-4, Nuclear Steam Supply System Safety Relief Valves and Electrical Backup Set Pressures, Capacities, and Duration of Blowdown, Rev. 19
Unit 2 FSAR Section 6.3.2, Emergency Core Cooling System Description, Rev. 20
Unit 2 FSAR Section 7.3.1, Emergency Core Cooling Systems Instrumentation and Control, Rev. 20

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TS 3.6.2.2, Suppression Pool Water Level
TS 3.5.1, ECCS - Operating
TS 3.3.5.1, Emergency Core Cooling System (ECCS) Instrumentation

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CR 2001003751, HPCI Thrust Bearing Oil Pressure Out of Spec, dated 05/14/01
CR 2002003058, No Open Indication Light When Valve 2E41-F104 Should Be Open, dated 03/28/02
CR 2002008733, Unit 2 HPCI Pump Inboard Bearing Oil Leak, dated 08/28/02
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CR 2001009819, Unit 2 HPCI System Main Pump Inboard Bearing and Turbine Outboard Bearing Oil Leaks, dated 10/26/01
CO 9904784, HPCI Pump Impeller Change, dated 07/07/99
CO 0006839, During HPCI Run Valve 2E41-F3040 Would Not Regulate Pressure, dated 08/14/00
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NRC IN 99-13, Insights from NRC Inspections of Low and Medium Voltage Circuit Breaker Maintenance Programs, dated July 26, 1999
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NRC IN 98-03, Inadequate Verification of Overcurrent Trip Setpoints in Metal-Clad Low Voltage Circuit Breakers, dated November 9, 1998
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NRC Augmented Inspection Team Report 50-321,366/00-01, dated February 28, 2000
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NRC Letter; Draft Safety Evaluation Report, Procedures Generation Package, Hatch Nuclear Plant Units ½, dated June 5, 1985
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