August 5, 2002

Mr. John L. Skolds
President and CNO
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
5th Floor
Warrenville, IL 60555

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION - NRC INSPECTION REPORT 50-277/02-011, 50-278/02-011

Dear Mr. Skolds:

On June 21, 2002, the NRC completed an inspection at the Peach Bottom Atomic Power Station. The enclosed report documents the inspection findings which were discussed on June 21, 2002, with Mr. Gordon Johnston and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety system design and performance capability of the residual heat removal (RHR) and high pressure coolant injection (HPCI) systems and compliance with the Commission's rules and regulations. The inspection consisted of a selected examination of calculations, drawings, procedures and records, observations of activities and interviews with personnel.

Based on the results of this inspection, the team identified two findings of very low safety significance (Green), one of which was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because the issue has been entered into your corrective action program, the NRC is treating this issue as a non-cited violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy, issued May 1, 2000, (65FR25368). If you contest this non-cited violation, 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 20555-0001, with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Peach Bottom 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/

Lawrence T. Doerflein, Chief
Systems Branch
Division of Reactor Safety

Docket No. 50-277,50-278
License No. DPR-44,DPR-56

Enclosure: Inspection Report 50-277/02-011 and 50-278/02-011

cc w/encl:
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REGION I

Docket No.	50-277, 50-278
Licensee No.	DPR-44, DPR-56
Report No.	50-277/02-011, 50-278/02-011
Licensee:	Exelon Generation Company, LLC Correspondence Control Desk 200 Exelon Way, KSA 1-N-1 Kennett Square, PA 19348
Facility:	Peach Bottom Atomic Power Station Units 2 and 3
Location:	1848 Lay Road Delta, Pennsylvania
Dates:	June 3 - 7 and June 17-21, 2002
Inspectors:	F. Arner, Senior Project Engineer S. Chaudhary, Reactor Inspector F. Jaxheimer, Reactor Inspector R. Moore, Reactor Inspector (Part time) B. Norris, Senior Reactor Inspector L. Scholl, Senior Reactor Inspector P. Wagner, USNRC Contractor
Approved by:	Lawrence T. Doerflein, Chief Systems Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000277-02-011, IR 05000278-02-011; Exelon Generation Company; on 06/03-06/21/2002; Peach Bottom Atomic Power Station; Units 2 and 3. Safety System Design and Performance Capability.

The inspection was conducted by five region I inspectors, one region II inspector (part time), and one NRC contractor. Two findings of very low safety significance (Green) were identified, one of which was considered a non-cited violation. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be “green” or be assigned a severity level after NRC management review. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

Cornerstone: Mitigating Systems

Green: The team identified a finding concerning an inadequate emergency operating procedure (EOP) for returning the suction of the high pressure coolant injection (HPCI) pump to the condensate storage tank (CST) to ensure the self cooled HPCI lube oil temperatures would remain within the analyzed limit. This issue was associated with the HPCI safety function during a postulated anticipated transient without scram.

The issue was considered to be of very low safety significance (Green) based on a Phase 1 evaluation of the SDP since there was no actual loss of the HPCI system, and was determined to be a non-cited violation (NCV) of the Peach Bottom Technical Specifications, Section 5.4.1.b., “Procedures.” (Section 1R21.1)

Green: The team identified that the HPCI and Reactor Core Isolation Cooling (RCIC) surveillance procedures incorporated steps which cycled 12 HPCI system valves and 8 RCIC valves, some several times, before the ASME in-service timing test. The team determined that this practice was unrecognized equipment preconditioning which had the potential to mask the as found condition of the valves.

The issue was determined to be a finding of very low safety significance (Green) based on a Phase 1 evaluation of the SDP because there was no actual loss of a valve safety function. (Section 1R21.2)

Report Details

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R21 Safety System Design and Performance Capability (IP 71111.21)

a. Inspection Scope

The team reviewed the design and performance capability of the residual heat removal (RHR) system and the high pressure coolant injection (HPCI) system. The significance determination process (SDP) worksheets and the individual plant examination (IPE) were reviewed to identify initiating events where these systems were credited with the capability of performing mitigating functions. Specific core damage accident sequences were selected in order to review the success criteria, including the required mission time for both the RHR and HPCI systems. The capability of HPCI to respond to a small break loss of coolant accident was reviewed. In addition, the HPCI function relative to a loss of offsite power initiating event was reviewed to ensure injection capability against a reactor vessel pressure corresponding to the setting of the lowest safety relief valves.

Furthermore, the plant risk assessment model credited HPCI as a viable injection source during a station blackout. For this scenario, the team reviewed the capability of the system to function without room cooling, with the suction of the pump aligned to the torus with increasing water temperature, while powered by the station batteries until the alternate source of power was assumed to be restored at one hour into the event. The team also reviewed the ability of the HPCI system to function in mitigating the anticipated transient without a scram (ATWS) initiating event. With respect to the RHR system, the team focused on specific modes of operation, including low pressure coolant injection (LPCI), containment (torus) cooling, and the capability of the high pressure service water pumps to perform late injection through the RHR piping for transients without the power conversion system.

The scope of the mechanical design review included: (1) a review of the HPCI turbine and governor controls including maintenance performed and test results for lubrication and control oil parameters; (2) a review of the performance of the Unit 2 and 3 HPCI booster and main pumps to ensure that the system would be able to provide the expected flow rate at the required pressure; (3) a verification that suction sources for the HPCI pump would be available during accident/special event conditions and the affect on lube oil cooling while taking suction from the torus; (4) a review of the technical adequacy of net positive suction head curves utilized in the emergency operating procedures for both the RHR and HPCI booster pumps; (5) a review of the technical adequacy of the maximum design basis differential pressures assumed for selected risk significant valves; and (6) a review of the B & C RHR pump runout protection and low pressure coolant injection loss of offsite power selection modification.

Additional mechanical design aspects reviewed included design documentation, drawings, HPCI operability determinations, calculations of RHR system capacity, RHR pump minimum flow and runout protection, and adequacy of the high pressure service water cross-tie capability for containment flooding. The impact on RHR net positive suction head (NPSH) due to the installation of the suction strainers in the torus was reviewed. The team reviewed the availability and reliability of the RHR room heating, ventilation, and air conditioning (HVAC) equipment to provide adequate equipment space environmental conditions during normal and accident conditions. This included a review of room heat load calculations for accident conditions, and performance history of the required equipment. The team performed field walkdowns of the accessible RHR piping and HPCI equipment for Unit 2 and 3 to assess the material condition and verify that the installed configuration was consistent with design drawings and design inputs to calculations. Additionally, the team reviewed the potential for common cause failure of the RHR pumps due to potential flooding in the equipment spaces.

The team reviewed the design and performance capabilities of the electrical and instrumentation and control systems to support the operation of the RHR and HPCI systems under accident and transient conditions. These reviews included verification that selected design requirements and commitments contained in the Updated Final Safety Analysis Report (UFSAR), design documents, and industry standards were being fulfilled. Documents reviewed included drawings, calculations (including instrument setpoint and loop uncertainty calculations), engineering analyses, accident analyses, work orders and hardware modifications. The team reviewed electrical testing and operating procedures to verify selected design parameters were being tested and that recommendations and restrictions contained in the vendor technical manuals for selected components had been incorporated. For example, the team reviewed the battery testing procedures against the technical specification requirements and IEEE Standards. Additionally, vendor information in the form of service information letters (SILs) were reviewed to ensure the licensee properly evaluated and incorporated applicable recommendations.

The team evaluated the adequacy of the circuit protection features and performed independent calculations and analyses to verify that the values utilized in the licensee's computer generated calculations were correct. The independent calculations included evaluations of circuit data based on conductor size and length and the type and resistance of the contractors and protective devices. The acceptability of the circuit breaker, fuse and thermal overload coordination related to selected system components was evaluated. The team also reviewed the direct current (DC) system voltage regulation to ensure adequate voltage levels were available at required loads under normal test and accident conditions. The adequacy of voltage supplied to the HPCI turbine controls was reviewed in detail.

Components selected for detailed review in the electrical area included the 125/250 volt batteries, the HPCI auxiliary oil pump (AOP), the HPCI steam admission valve (MO-14), and the RHR pump motors. To ensure the capability of the RHR pump motors the team reviewed emergency diesel generator (EDG) loading and evaluated the circuitry utilized to accomplish load stripping along with applicable procedures to ensure the circuitry was being adequately tested. The HPCI turbine AOP and MO-14 circuit coordination calculations were evaluated and selected inputs were verified by independent calculation.

The team reviewed the procedures used to operate and test the RHR and HPCI systems during normal and accident conditions. The types of procedures reviewed included: system operating procedures, abnormal and emergency operating procedures, alarm response cards, and surveillance tests. In particular, the impact of power re-rate on system design margins was reviewed for both the RHR and HPCI systems to verify the adequacy of acceptance criteria in system testing. The team reviewed the training lesson plans for the systems to ensure they appropriately described the design features of the systems.

The team reviewed the station blackout (SBO) procedures with respect to the assumption of placing an RHR pump in service within one hour of the blackout condition. The review included the original design, modifications related to adding the 3EA transformer as an option during maintenance, and the associated procedures. The inspector also walked down the SBO lineup in the plant, at the SBO switchyard, and at the Conowingo Hydro-Electric station to verify the capability of placing an RHR pump back in torus cooling within the one hour time-frame assumed in SBO analyses.

The team selected a sample of condition reports and action requests associated with the selected systems to verify the licensee was identifying and correcting design issues at an appropriate threshold, entering them in the corrective action program, and taking appropriate corrective actions. Documents reviewed and personnel interviewed during the inspection are listed in Attachment A.

b. Findings

.1 High Pressure Coolant Injection Function- ATWS Analyses

Introduction

The inspection team identified a finding concerning an inadequate emergency operating procedure (EOP) for returning the suction of the high pressure coolant injection pump to the condensate storage tank (CST) to ensure the self cooled HPCI lube oil temperatures would remain within the analyzed limit. This issue was associated with the HPCI safety function during a postulated anticipated transient without scram. The issue was considered to be of very low safety significance (Green) since there was no actual loss of the HPCI system, and was determined to be a non-cited violation (NCV) of the Peach Bottom Technical Specifications, Section 5.4.1.b., “Procedures.”

Description

During a review of the power re-rate analysis along with discussions with licensee personnel, the team noted that an assumption in the anticipated transient without a scram scenario was that the suction for the HPCI pump was always from the CST. The reason to maintain the suction from the preferred CST source for as long as possible, in the event of a main steam isolation valve (MSIV) closure ATWS, is because of the increasing temperature of the torus water. During this ATWS, the safety relief valves (SRVs) lift due to high pressure in the reactor vessel; the SRVs discharge to the torus (also referred to as the suppression chamber). The torus water was calculated in the licensee’s plant specific re-rate analysis to be as high as 188 degrees Fahrenheit. The HPCI pump is self-cooled, and the lube oil for the control system, pump and turbine bearings had been analyzed for suction water temperatures of up to 180°F.

The team reviewed the TRIP procedures (Transient Response Implementation Plan - Peach Bottom's term for the EOPs) to determine how this assumption was translated into the TRIPs. The team noted that the TRIP procedure for an ATWS (T-117, "Level/Power Control") directed the use of HPCI, with the statement -"CST SUCTION IS PREFERRED, DEFEAT HIGH TORUS LEVEL SWAP OVER USING T-226 IF NECESSARY." Secondary TRIP procedure T-226, "Defeating HPCI High Torus Level Suction Transfer," assumed that the swap over from the CST to the torus had not already occurred. During interviews with the HPCI system engineer, the Operations Support Manager, and the EOP Program Manager, the team determined that the licensee personnel thought that there was sufficient time to defeat the swap over.

The inspectors questioned whether there was enough time for the operators to implement T-226 in the event of an ATWS. Subsequently, the licensee ran several ATWS scenarios in the simulator and found that an automatic swap over due to increasing torus level occurred between 3 and 9 minutes; i.e., before the operators would be directed to implement T-226. Although T-117 contained guidance for defeating the swap over, there was no guidance to switch the HPCI pump suction back to the CST if it had already automatically transferred to the torus on high torus water level. The licensee initiated a condition report and planned to revise T-226 to incorporate the steps for returning the HPCI suction to the CST. The inspectors considered the planned action to be reasonable.

Analysis

The lack of appropriate procedural direction to maintain HPCI on its preferred suction source was considered to be more than minor because the increased suction temperature could affect the availability and reliability of the HPCI system, a mitigating system of the reactor safety cornerstone; specifically, the finding was associated with the procedure quality attribute associated with a mitigating system and affected the objective of ensuring the capability of the HPCI system. The issue was screened green in phase 1 of MC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." This issue was determined to be of very low safety significance (Green) because HPCI would have remained available following an ATWS event. The conditions where HPCI may have become unavailable following an ATWS were based on a "worst case" analysis that assumed adverse conditions that were not currently applicable (i.e 5% of the RHR heat exchanger tubes plugged and fouling factors at design values). Additionally, the frequency of ATWS events is very low and even if HPCI was unavailable, operators could still provide vessel makeup by depressurizing the reactor and injecting with low pressure pumps.

Enforcement

Technical specification (TS) 5.4.1.b requires that written procedures shall be established, implemented, and maintained covering emergency operating procedures. Contrary to the above, the TRIP procedures were inadequate in that procedural direction did not ensure the continued operation of the HPCI system consistent with plant specific analysis assumptions (i.e., ensuring that the HPCI pump would always take a suction from the CST during an MSIV closure ATWS condition by either defeating the high torus water level swap or by giving adequate guidance to return the suction to the CST after it had swapped). This was determined to be a violation of TS 5.4.1.b. This issue was associated with an inspection finding that was characterized by the Significance Determination Process as having very low risk significance (i.e., Green) and is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy. This issue is in the licensee's corrective action program as Condition Report CR-112172. **(NCV 50-277;278/02-011-01)**

.2 High Pressure Coolant Injection Surveillance Testing

Introduction

The team identified that HPCI and Reactor Core Isolation Cooling (RCIC) surveillance procedures incorporated steps which cycled 12 HPCI system valves and 8 RCIC valves, some several times, before the ASME in-service timing test. The team determined that this practice had the potential to mask the as found condition of the valves. The team noted that the valve test sequence was not in accordance with the definition of acceptable preconditioning identified in Inspection Manual Part 9900 technical guidance (Maintenance-Preconditioning of Structures, Systems, And Components Before Determining Operability), which was referenced in Inspection Procedure (IP) 71111.22, "Surveillance Testing." The issue was determined to be a finding of very low safety significance (Green) because there was no actual loss of a valve safety function.

Description

During a review of surveillance testing procedures ST-0-023-301-2/3, "HPCI Pump, Valve Flow & Unit Cooler Functional & IST," for proper acceptance criteria, the team noted that the HPCI procedure cycled 12 valves before the ASME In service timing test. The observation was compared to information notice (IN) 97-16, "Preconditioning Of Plant Structures, Systems, And Components Before ASME Code In service Testing Or Technical Specification Surveillance Testing," and NRC Inspection Manual (IM) Part 9900 guidance on preconditioning. The surveillance activity matched the definition of preconditioning of components which was; "the alteration, variation, manipulation, or adjustment of the physical condition of a component before technical specification surveillance or ASME Code testing." The basis for the valve exercising sequence in the surveillance procedures had not been evaluated for the effects of preconditioning and did not meet the conditions for acceptable preconditioning defined in technical guidance document IM Part 9900. The team found that both air operated and motor operated valves in the HPCI and RCIC systems were cycled prior to the ASME time testing. The team noted that cycling of air operated valves prior to stroke time testing can bypass or mask the as-found condition of these components. The team was concerned that motor operated stroke times may also change after initial cycles of the valves for standby systems. The relevant technical issues include pressure locking, mechanical drag when clearing the backseat, operator gearbox lubrication, and valve stem lubrication. The team noted that direct current motor speed can be affected by the above issues and therefore the as-found condition of the valves may be masked by the current test sequence for the HPCI and RCIC systems.

The licensee indicated that procedure changes which created the testing sequence were made in response to a related significant operational experience report 81-13 and GE SIL #336, “Surveillance testing recommendations for HPCI & RCIC systems,” which discussed preventing preconditioning of the HPCI and RCIC turbines. However, the team determined that the majority of valves identified do not have the potential to influence turbine startup testing as discussed in the referenced documents, if they were stroke timed prior to the turbine runs. The valves included the HPCI gland seal condenser condensate pump discharge line to radwaste, the CST suction valves, torus suction valves, minimum flow line valves, and the test return valves. The licensee’s test sequence was not identified as a violation of a specific ASME code testing requirement. However, IM Part 9900 has a documented position on preconditioning as it relates to the wording of 10 CFR50.55a regarding operational readiness.

Analysis

Appendix E of Manual Chapter 0612 was not applicable with regard to this finding. The unrecognized valve preconditioning was determined to be more than minor because it was associated with the procedure quality attribute for the HPCI and RCIC mitigating systems and affected the mitigating system objective to ensure the availability and reliability of the applicable valves. The finding was associated with testing performed to determine the operability and reliability of the HPCI and RCIC systems and therefore was processed by Manual Chapter 0609, the significant determination process (SDP). The team reviewed available valve data, including design margin, preventive maintenance history and actual historical valve test results and concluded that there was no indication that the associated valves could not support their safety functions. The issue was determined to be a finding of very low safety significance (Green) through a phase 1 SDP review because there was no actual loss of a safety function of a system.

Enforcement

No violation of regulatory requirements occurred. Exelon entered this issue into the corrective action system as Condition Report (CR) # 00111936. **(FIN 50-277;278/02-011-02)**

4. OTHER ACTIVITIES (OA)

4OA6 Exit Meeting Summary

The team presented the inspection results to Mr. Gordon Johnston and other members of the licensee’s staff at an exit meeting on June 21, 2002. Proprietary information examined during the inspection was identified and returned to the licensee at the conclusion of the inspection.

ATTACHMENT 1

SUPPLEMENTARY INFORMATION

Key Points Of Contact

Exelon Generation Company

M. Alfonso, Director Training
J. Armstrong, Nuclear Oversight Manager
C. Behrend, Branch Manager-NSSS
P. Davison, Site Engineering Director
M. Delowery, Senior Manager Plant Engineering
D. Falcone, Operations Support Manager
B. Hanson, Operations Director
J. Heyne, Maintenance Support Manager
G. Johnston, Plant Manager
J. Jordan, Manager Mechanical Design
T. LaMontange, Reactor Operator
J. Lyter, EOP Program Manager
J. Pomeroy-Senior Reactor Operator
D. Warfel, Senior Manager Design
J. Zardus, HPCI System Engineer

United States Nuclear Regulatory Commission

M. Buckley	Resident Inspector
L. Doerflein	Chief, Systems Branch, R1 DRS
A. McMurtray	Senior Resident Inspector

List of Items Opened, Closed, and Discussed

Opened/Closed

50-277;278/02-011-01	NCV	Trip Procedures Inconsistent With Plant Specific Analysis
50-277;278/02-011-02	Finding	Preconditioning of HPCI, RCIC Valves prior to IST

LIST OF DOCUMENTS REVIEWED

PROCEDURES:

A-C-1, App 3, Exh 11	Preparation of Rapid Response Cards, Revision 0
A-C-226	TRIP & SAMP Procedures Program, Revision 1
A-C-226-01	TRIP & SAMP Procedures Writer's Guide, Revision 2
AO-32.2-2	HPSW Injection into the Reactor Vessel, Revision 1
GP-2	Normal Plant Start-Up, Revision 99
GP-3	Normal Plant Shutdown, Revision 90
HU-AA-104-101	Procedure Use & Adherence, Revision 0
LS-AA-105	Operability Determinations, Revision 0
LS-AA-125	Corrective Action Program (CAP) Procedure, Revision 2
NOM-C-10.2	Operations Section Performance Standards (OSPS) Introduction and Overview, Revision 0
NOM-C-7.1	Procedure Use, Revision 2
OP-PB-108-101-1001	Simple Quick Acts / Transient Acts, Revision 0
PLOR-00-05C	Training Material: Station Blackout Modification P00907, Revision 0
PLOR-087P	Training Material: Defeating HPCI High Level Torus Level Suction Transfer, Revision 11
PLORT-02-01B	Training Material: Summer Readiness, Revision 0
PLOT-1555	Training Material: Special Events (SE), Revision 5
PLOT-2111	Training Material: T-111, Level Restoration, Revision 0
PLOT-5051	Training Material: Substations, Revision 2
PNLO-3115	Training Material: T-200 & T-300 Trip Procedures, Revision 1
PNLOC-00-03C	Training Material: Mod P00907 Susquehanna 351/191 Distribution Line Enhancements, Revision 1
PNLOC-00-06B	Training Material: LOOP (Back Feeding and Attachments W & Z), Revision 0
PSEG-0215R	Training Material: ATWS [T-117], Revision 14
PSEG-0417R	Training Material: LOOP with no DGs Available, Revision 2
PSEG-0514L	Training Material: SE-11.1, Revision 0
PSTG-A-Cautions	Operator Precautions - Appendix A, Revision 3
PSTG-A-Intro	Introduction, Revision 1
PSTG-A-T-101	RPV Control Guideline, Revision 7
PSTG-A-T-117	Contingency #5 Level/Power Control, Revision 9
PSTG-B-Cautions	Operator Precautions - Appendix B, Revision 5
PSTG-B-Intro	Introduction, Revision 1
PSTG-B-T-101 RC/Q	RPV Control Guideline, Revision 7
PSTG-B-T-101 RC/RL	RPV Control Guideline, Revision 8
PSTG-B-T-101 RC/P	RC Guideline Part RC-P - Appendix B, Revision 5
PSTG-B-T-117	Contingency #5 Level/Power Control, Revision 10
PSTG-Cautions	Operator Precautions, Revision 2
PSTG-T-101	RPV Control Guideline, Revision 6
PSTG-T-117	Contingency #5 Level/Power Control, Revision 8
RRC-10.1-2	RHR System Torus Cooling During a Plant Event, Revision 0
RRC-10.2-2	RHR System LPCI Manual Start During a Plant Event, Revision 0
RRC-11.1-2	Standby Liquid System Initiation During a Plant Event, Revision 0
RRC-13.1-2	RCIC System Operation During a Plant Event, Revision 0

RRC-14,1-2	Core Spray Manual Initiation During a Plant Event, Revision 0
RRC-16.1-2	Bypass & Restore Instrument N_2 Supply to Drywell, Revision 0
RRC-1G.1-2	Automatic Depressurization System, Revision 0
RRC-1G.2-2	Relief Valve Manual Operation During a Plant Event, Revision 1
RRC-23.1-2	HPCI System Operation During a Plant Event, Revision 2
RRC-3B.1-2	Alternate Rod Injection During a Plant Event, Revision 0
RRC-44A.1-2	Maximize Drywell Cooling, Revision 2
RRC-53.1-2	Unit 2 House Loads Transfer During a Plant Event, Revision 0
RRC-55.1-2	Cross-Tie of 480V Load Centers During a Plant Event, Revision 0
RRC-7J.1-2	Drywell & Torus H_2/O_2 Sampling Startup During a Plant Event - CAD Mode, Revision 0
RRC-94.1-2:1	URO Scram Reports, Revision 0
RRC-94.1-2	Reactor Operator Scram Actions, Revision 0
RRC-94.2-2	Plant Reactor Operator Scram Actions, Revision 0
RRC-94.2-2:1	PRO Scram Reports, Revision 0
SAMP-1 Bases	RPV & Primary Containment Flooding Control, Revision 1
SE-1	Plant Shutdown from the Remote Shutdown Panel, Revision 16
SE-1 Bases	Plant Shutdown from the Remote Shutdown Panel, Revision 16
SE-10	Alternate Shut Down, Revision 11
SE-10 Att 9	HPCI Operations from the Alternative Shutdown Panel, Revision 1
SE-10 Bases	Plant Shutdown from the Alternative Shutdown Panels, Revision 12
SE-11 Bases	Loss of Off-Site Power, Revision 11
SE-11	Loss of Off-Site Power, Revision 12
SE-11.1	Operating Station Blackout Line During a LOOP Event, Revision 3
SO-10.1.A-2	RHR System Set-Up for Automatic Operation, Revision 3
SO-10.1.A-2A COL	RHR System Set-Up for Automatic Shutdown, Revision 18
SO-10.1.A-2B COL	RHR System Set-Up for Automatic Shutdown, Revision 13
SO-10.1.B-2	RHR System Shutdown Cooling Mode Manual Start, Revision 28
SO-10.1.C-2	RHR System Precise Reactor Temperature Control, Revision 4
SO-10.1.D-2	RHR System Torus Cooling, Revision 15
SO-10.2.A-2	RHR System LPCI Shutdown & Return to Standby, Revision 2
SO-10.2.B-2	RHR Shutdown Cooling Mode Shutdown, Revision 17
SO-10.3.A-2	RHR System A Loop Filling & Venting, Revision 11
SO-10.3.C-2	Manually Venting of the RHR LPCI & Containment Spray Line Vent Accumulator Lines, Revision 2
SO-10.3-2	RHR System Fuel Pool Cooling Mode, Revision 3
SO-10.5.A-2	RHR System Piping Flush, Revision 2
SO-10.7.B-2	RHR System Automatic Response During LOCA and Manual System Initiation upon Automatic Injection Failure, Revision 6
SO-10.7.D-2	RHR Shutdown Cooling Operation Through MO-2-10-020 “RHR Loop X-Tie”, Revision 0
SO-10.8.A-2	RHR System Routine Inspection, Revision 2
SO-14A.1.A-2	Torus Water Cleanup and Level Control, Revision 8
SO-23.1.A-2	HPCI System Setup for Automatic or Manual Operation, Revision 10
SO-23.1.B-2	HPCI System Manual Operation, Revision 15
SO-23.2.A-2	HPCI System Shutdown, Revision 14
SO-23.7.A-2	HPCI System Automatic Initiation Response, Revision 7
SO-23.7.B-2	Transfer of HPCI Pump Suction from CST to Torus, Revision, 4

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ST-M-57B-733-3	3C Battery Performance Test conducted 9/30/99
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9970-613	HPCI Diagram Wiring Schematic

Walworth Valve Dwg A-4088-M-23A, Cast Steel Globe Valve
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M-1-EE-222	Sh. 2 Connection Diagram Panel 9-48, Rev. 34
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PE-0155	Seismic Evaluation of Battery Racks, Rev. 1
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PE-0196	125/250 Volt DC System Coordination, Rev. 2
PE-0155	Seismic Evaluation of Battery Racks
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PM-1013	Minimum Containment Pressure Available, Rev. 3
ME-507	acceptance criteria for the ST of RHR pumps to meet the TS section 4.5.A.d
PM-846	Elimination of ESE Flow requirements to RHR pumps
PM-958	Calculate RHR/CS room temperature post LOCA
MISLCalc 64	Verification of ECCS Strainer Pressure Drops for PB Units 2 and 3
ME-293	Pressure Drop for HPCI Injection at Flow rate of 5000 gpm, Rev. 0
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ME-502	# tubes allowed plugged in unit 3 HPCI lube oil cooler
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VENDOR TECHNICAL MANUALS

M-1-JJ-30	Terry Turbine Manual
E-5-166	Fairbanks-Morse Vendor Manual
E-5-167	Operation and maintenance manual for EDGs
E-13-123	Exide Instructions, installing and operating station batteries

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ECR 93-03983, Replace RCIC Flow Controller FC-2-13-091
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70324	98027	112172	198915	199124
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A1243766	A1367719	A1328987	A1344638	A1211046
A1246921	A1269128	A1332748	A1350942	A1373248
A1253678	A1274064	A1332921	A1365067	A1238474
A1263593	A1310094	A1333654	A1369351	A1348993
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WORK ORDERS

R0547635
R0547879
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R0645522
R0744366
R0477376
NCR PB 97-02609 001
NCR-PB-94-00005
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List Of Acronyms

AOP	Auxiliary Oil Pump
ATWS	Anticipated Transient Without Scram
CST	Condensate Storage Tank
DBA	Design Basis Accident
DBD	Design Bases Document
DC	Direct Current
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
HPCI	High Pressure Coolant Injection
HPSW	High Pressure Service Water
HVAC	Heating, Ventilation & Air Conditioning
IPE	Individual Plant Examination
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
MSIV	Main Steam Isolation Valve
NPSH	Net Positive Suction Head
PBAPS	Peach Bottom Atomic Power Station
PECO	Peco Energy
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
SAMP	Severe Accident Management Plan
SBO	Station Black-Out

SDP	Significance Determination Process
SIL	Service Information Letter
SRV	Safety Relief Valve
TRIP	Transient Response Implementation Plan
UFSAR	Updated Final Safety Analysis Report
USNRC	United States Nuclear Regulatory Commission
V&V	Verification & Validation