From: Charles R. Ogle, R2
To: Eng Branch
Date: 1/16/04 10:40AM
Subject: PEER REVIEW OF INSPECTION REPORTS

At the branch meeting on Feb 25th, I'd like to conduct a peer review of the attached inspection reports. We will focus exclusively during this peer review on the 4 part writeups developed in both of these reports.

Please prepare for this branch meeting by reviewing the 4-part writeups against the criteria contained in MC 0612 http://nrr10.nrc.gov/IM/manualchapters.html

and the criteria contained in the handout I have put in each of your boxes.

Norm will serve as the moderator for this discussion and prepare a summary of the Branch's observations. Thanks.

H/10

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Subject: PEER REVIEW OF INSPECTION REPORTS
Creation Date: 1/16/04 10:39AM
From: Charles R. Ogle

Created By: CRO@nrc.gov

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REGION II
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ATLANTA, GEORGIA 30303-8931

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November 18, 2003

EA-00-022
EA-01-310

Carolina Power & Light Company
ATTN: Mr. James Scarola
Vice President - Harris Plant
Shearon Harris Nuclear Power Plant
P. O. Box 165, Mail Code: Zone 1
New Hill, North Carolina 27562-0165

PROPRIETARY INFORMATION REMOVED

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT - NRC FIRE PROTECTION INSPECTION REPORT NO. 05000400/2003007

Dear Mr. Scarola:

On October 21, 2003, the U.S. Nuclear Regulatory Commission (NRC) completed an in-office review of the significance of the triennial fire protection inspection findings of inspection report 05000400/2002011 related to your Shearon Harris Nuclear Power Plant. The enclosed report documents the results of our significance determination, which was discussed on October 21, 2003, by telephone with Mr. R. Duncan and other members of your staff.

This report documents two NRC-identified findings of very low significance (Green). Both of these findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these two findings as non-cited violations (NCVs) consistent with Section VI.A. of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Shearon Harris Nuclear Power Plant.

CP&L

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In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, portions of its enclosure and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of

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NRC's document system (ADAMS). However, the NRC is continuing to review the appropriate classification of the Phase 3 significance determination process analysis (Attachment 2) within our records management program, considering changes in our practices following the events of September 11, 2001. Using our interim guidance, the attached analysis has been marked as Proprietary Information or Sensitive Information in accordance with Section 2.790(d) of Title 10 of the Code of Federal Regulations. Please control the document accordingly (i.e., treat the document as if you had determined that it contained trade secrets and commercial or financial information that you considered privileged or confidential). We will inform you if the classification of these documents change as a result of our ongoing assessments. ADAMS is accessible from the NRC Website 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 No.: 50-400
License No.: NPF-63

Enclosure: Inspection Report 05000400/2003007
w/Attachments: 1. *Supplemental Information*
2. Phase 3 SDP Analysis (Contains Proprietary Information)

cc w/encl and Attachment 1:
James W. Holt, Manager
Performance Evaluation and
Regulatory Affairs CPB 9
Carolina Power & Light Company
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Robert J. Duncan II
Director of Site Operations
Carolina Power & Light Company
Shearon Harris Nuclear Power Plant
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(cc w/encl and Attachment 1 cont'd - See page 3)

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(SEE PREVIOUS PAGE FOR CONCURRENCES)

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NAME	RSchin	WRogers	KO'Donohue	PFredrickson	CEvans		
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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.: 50-400

License No.: NPF-63

Report No.: 05000400/2003007

Licensee: Carolina Power & Light (CP&L)

Facility: Shearon Harris Nuclear Power Plant

Location: 5413 Shearon Harris Road
New Hill, NC 27562

Dates: February 1, 2003 - October 21, 2003

Inspectors: W. Rogers, Senior Reactor Analyst, Region II
R. Schin, Senior Reactor Inspector, Region II

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

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Enclosure

SUMMARY OF FINDINGS

IR 05000400/2003-007; 02/01/2003 - 10/21/2003; Shearon Harris Nuclear Power Plant; Significance Determination of Fire Protection Findings.

The in-office review was conducted by a regional inspector, a regional senior reactor analyst, and NRC Headquarters risk analysts. Two Green findings, each a non-cited violation (NCV), were identified. The significance of issues is indicated by their color (Green, White, Yellow, Red) using IMC 0609 "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be "Green" or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector Identified Findings

Cornerstones: Mitigating Systems and Initiating Events

- Green. The inspectors identified a non-cited violation (NCV) of Operating License Condition 2.F, the Fire Protection Program, and Technical Specification 6.8.1, Procedures and Programs, for inadequate implementation of the fire protection program. Physical and procedural protection for equipment that was relied on for safe shutdown (SSD) during a fire in fire safe shutdown analysis (SSA) areas 1-A-BAL-B-B1, 1-A-BAL-B-B2, 1-A-BAL-B-B4, 1-A-BAL-B-B5, 1-A-EPA, and 1-A-BAL-C of the reactor auxiliary building was inadequate. Consequently, a fire in one of these SSA areas could result in a reactor coolant pump seal loss of coolant accident event, a main steam power-operated relief valve failed open event, a loss of high pressure safety injection, and/or a loss of component cooling water to the reactor coolant pump seals. Licensee corrective action included assigning an additional operator to be available to perform post-fire SSD actions and performing a complete review of the SSA and related operating procedures.

  This finding was greater than minor because it involved a lack of required fire barriers for equipment that was relied upon for safe hot shutdown following a fire. The finding also had more than minor safety significance because it affected the objectives of the Mitigating Systems and Initiating Events Cornerstones. The finding affected the availability and reliability of systems that mitigate initiating events to prevent undesirable consequences and also affected the likelihood of occurrence of initiating events that challenge critical safety functions. The finding was of very low significance (Green) because of the low fire ignition frequencies, lack of combustible materials in critical locations, and the effectiveness of the fire protection features and the unaffected SSD equipment to mitigate a fire in each of the affected fire zones/areas.
  [Section 1R05.1.b.(1)]

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- Green. The inspectors identified a non-cited violation (NCV) of Operating License Condition 2.F, the Fire Protection Program, and Technical Specification 6.8.1, Procedures and Programs, for inadequate corrective action for previous Violation 50-400/02-08-01. Corrective action for that violation had included creating a new auxiliary control panel fire area (1-A-ACP) in 2002. However, that corrective action was not adequate because physical and procedural protection for equipment that was relied on for safe shutdown (SSD) during a fire in the new fire area was inadequate. Consequently, a fire in area 1-A-ACP could result in a loss of auxiliary feedwater and a main steam power-operated relief valve failed open event. Licensee corrective actions in response to this finding included assigning an additional operator to be available to perform post-fire SSD actions and performing a complete review of the SSA and related operating procedures.

  This finding was greater than minor because it involved inadequate fire barriers for equipment that was relied upon for safe hot shutdown following a fire. The finding also had more than minor safety significance because it affected the objectives of the Mitigating Systems and Initiating Events Cornerstones. The finding affected the availability and reliability of systems that mitigate initiating events to prevent undesirable consequences and also affected the likelihood of occurrence of initiating events that challenge critical safety functions. The finding was of very low significance (Green) because of the very low ignition sources in the fire area, manual suppression capability, and the power conversion system not being affected by a fire in this fire area. [Section 1R05.1.b.(2)]

B. Licensee-Identified Violations

None

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REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events and Mitigating Systems

1R05 FIRE PROTECTION

.1 Significance Determination for Triennial Fire Protection Inspection Findings

a. Inspection Scope

In Inspection Report (IR) 50-400/02-11, nine findings were identified as unresolved items (URIs) pending completion of the NRC significance determination process (SDP). The nine URIs were as follows:

- URI 50-400/02-11-01, Failure to Protect Charging System MOV 1CS-165, VCT Outlet to CSIPs, From Maloperation Due To a Fire
- URI 50-400/02-11-02, Failure to Protect Charging System MOVs 1CS-169, 1CS-214, 1CS-218, and 1CS-219 From Maloperation Due To a Fire
- URI 50-400/02-11-03, Failure to Protect Charging System MOVs 1CS-166, 1CS-168, and 1CS-217 From Maloperation Due To a Fire
- URI 50-400/02-11-04, Failure to Protect Component Cooling MOVs 1CC-251 and 1CC-208, CC for RCP Seals, From Maloperation Due To a Fire
- URI 50-400/02-11-05, Reliance on Manual Actions in Place of Required Physical Separation or Protection From a Fire
- URI 50-400/02-11-06, Fire SSD Operator Actions With Excessive Challenges
- URI 50-400/02-11-07, Too Many Fire SSD Actions for Operators to Perform
- URI 50-400/02-11-08, Using the Boric Acid Tank Without Level Indication
- URI 50-400/02-11-09, Failure to Provide Required Emergency Lighting for SSD Operator Actions

This IR documents the results of the in-office completion of the NRC SDP with respect to the nine URIs. The significance determination was accomplished as described in NRC Inspection Manual Chapter (IMC) 0609, Signification Determination Process; IMC

0609A, Significance Determination of Reactor Inspection Findings for At-Power Situations; and IMC 0609F, Determining Potential Risk Significance of Fire Protection and Post-Fire Safe Shutdown Inspection Findings. This involved evaluating the significance of a potential fire in each of seven affected fire safe shutdown analysis (SSA) areas using the Phase 2 SDP, considering all examples of the findings that could be involved in each fire. To better assess the overall significance of all of the performance deficiencies, they were processed through the SDP as two overall findings: 1) Inadequate Implementation of the Fire Protection Program (FPP) for safe shutdown (SSD); and 2) Inadequate Corrective Action for a Previous White Fire Protection Finding.

In addition, the performance deficiencies which could result in the loss of a safety function were evaluated by NRC Headquarters risk analysts using the Phase 3 portion of the SDP. Included in this evaluation were extensive walkdowns of the applicable fire SSA areas by two NRC fire protection contractors to observe ignition sources and possible fire propagation pathways from these ignition sources that could affect the unprotected cables of concern. Also, electrical circuit drawings and the latest information on cable hot short failure mechanisms and probabilities were used to develop cable failure probabilities that could cause a loss of function for the unprotected cables of concern.

b. Findings

(1) Inadequate Implementation of the FPP for SSD

Introduction: An overall finding was identified in that the implementation of the FPP was inadequate. Eight of the nine URIs described in IR 50-400/02-11 were considered to include performance deficiencies related to this overall finding. Based on evaluating those performance deficiencies for their effects during fires that could occur in each of six (of the seven total) affected fire SSA areas, this overall finding was determined to have a very low significance (Green).

Description: The licensee's implementation of the FPP for ensuring the ability to safely shut down the plant during a fire was inadequate, in that:

- The fire SSA failed to identify several cables that were relied upon for SSD during a fire. Consequently, those cables were not provided with the required protection from fire damage. A fire could cause hot shorts in the cables which would result in maloperation of equipment that was relied upon for SSD during that fire.
- The SSA identified many cables that were relied upon for SSD during a fire, for which the licensee generally failed to provide the required physical protection from fire damage. Instead, the SSA designated that operator actions would be

taken to prevent or mitigate the effects of the fire damage. However, the licensee did not obtain NRC approval for these deviations from the approved FPP.

- Some of the operator actions that were designated by the SSA were not incorporated into operating procedures for SSD. Also, the operator actions in procedures differed in many respects from the operator actions that were analyzed in the SSA. For example, the operating procedures directed operators to use some different flowpaths than those analyzed in the SSA.

- Some operator actions in the SSD procedures would not work. They were too challenging, involved entering the area of the fire, were not adequately analyzed, or were too numerous for the available SSD non-licensed operator to perform.

Detailed examples related to this overall finding were included in the following eight URIs: 50-400/02-11-01, -02, -03, -04, -05, -07, -08, and -09.

Analysis: The inspectors and analysts evaluated the effects of the multiple examples of this overall finding during a fire that could occur in each of the six affected fire SSA areas of the reactor auxiliary building (RAB) using Phase 2 and Phase 3 of the SDP. Based on that analysis, the inspector and analysts concluded that this finding had more than minor safety significance because it involved a lack of required fire barriers for equipment that was relied upon for safe hot shutdown following a fire. The finding also had more than minor safety significance because it affected the availability and reliability objectives and the equipment performance attribute of the Mitigating Systems Cornerstone. In addition, it affected the Initiating Events Cornerstone in that it affected the objective of limiting the likelihood of occurrence of initiating events that challenge critical safety functions and also affected the design control attribute. The overall finding did not have more than very low safety significance (Green) because of the low fire ignition frequencies that could impact the cables of interest, the lack of combustible materials in critical locations, and the effectiveness of the fire protection features and the unaffected SSD equipment to mitigate a fire in each of the affected fire zones/areas.

Enforcement: As described in IR 50-400/02-11, Operating License Condition (OLC) 2.F requires that the licensee implement and maintain in effect all provisions of the FPP as described in the Final Safety Analysis Report (FSAR). The Updated FSAR (UFSAR), Section 9.5.1, FPP, states that outside containment, where cables or equipment (including associated non-essential circuits that could prevent operation or cause maloperation due to hot shorts, open circuits, or shorts to ground) of redundant SSD divisions of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, one of the redundant divisions must be ensured to be free of fire damage. Section 9.5.1 further states that if both divisions are located in the same fire area, then one division is to be physically

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protected from fire damage by one of three methods: 1) a three-hour fire barrier, 2) a one-hour fire barrier plus automatic detection and suppression, or 3) a 20-foot separation with no intervening combustibles and with automatic detection and suppression. The licensee had not received NRC approval for deviating from these requirements.

Also, OLC 2. F. and UFSAR Section 9.5.1 state that Branch Technical Position (BTP) 9.5-1 was used in the design of the FPP for safety-related systems and equipment and for other plant areas containing fire hazards that could adversely affect safety-related systems. BTP 9.5-1, Section C.5.g, "Lighting and Communication," paragraph (1), requires that fixed self-contained lighting consisting of fluorescent or sealed-beam units with individual eight-hour-minimum battery power supplies should be provided in areas that must be manned for SSD and for access and egress routes to and from all fire areas.

In addition, Technical Specification 6.8.1, Procedures and Programs, requires procedures as recommended by Regulatory Guide 1.33 and procedures for FPP implementation. Regulatory Guide 1.33 recommends procedures for combating emergencies, including fires. The licensee's interpretation of the FPP was that they could and would rely on proceduralized operator actions in place of physically protecting electrical cables for SSD equipment from fire damage. The operator actions were contained in Procedure AOP-36, Safe Shutdown Following a Fire, Rev. 21.

Contrary to the above requirements, the licensee failed to adequately implement and maintain in effect all of the provisions of the approved FPP. The licensee failed to ensure that one of the redundant SSD divisions of systems necessary to achieve and maintain cold shutdown conditions was protected from fire damage; failed to have adequate procedures for combating fire emergencies; and failed to provide the required emergency lighting in areas that must be manned for SSD; as described above in the eight examples of this overall finding. These conditions were identified by the NRC in December 2002 and had been in place for years. Because the identified examples of this failure to adequately implement and maintain in effect all of the provisions of the approved FPP are of very low safety significance and have been entered into the corrective action program [Action Requests (ARs) 76260, 80212, 80089, 69721, 80215, 75065, and 79047], this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 50-400/03-07-01; Inadequate Implementation of the Fire Protection Program for Safe Shutdown.

(2) Inadequate Corrective Action for a Previous White Fire Protection Finding

Introduction: In IR 50-400/02-08, the NRC left Violation 50-400/02-08-01 open for further NRC review of the new manual operator actions that were added for the new 1-A-ACP fire area, as part of the licensee's corrective action for the violation. In IR 50-400/02-11, the NRC documented the review of those new manual operator actions and identified that the licensee's corrective actions contributed to four new findings. For this significance determination, those findings were grouped into one overall finding of inadequate corrective action for a previous White fire protection finding. Based on evaluating the multiple examples of this overall finding for their effects during a fire that could occur in the new 1-A-ACP fire area, this overall finding was determined to have a very low significance (Green).

Description: The licensee's corrective actions for a previous White fire protection finding (Violation 50-400/02-08-01), associated with a Thermo-Lag fire barrier assembly between the 'B' train switchgear room / auxiliary control panel and the 'A' train cable spreading room, were inadequate. The corrective actions were inadequate because they failed to rectify deficiencies in design, construction, and operation related to SSD from a fire in the new 1-A-ACP fire area. Consequently, a fire in area 1-A-ACP could result in a loss of auxiliary feedwater and a main steam power-operated relief valve failed open event. The licensee's corrective actions contributed to four new findings that are now grouped into the overall finding of inadequate corrective action:

- The corrective actions created a new fire area (1-A-ACP) and many new manual operator actions for a fire in the new fire area instead of providing the required physical protection of cables. This finding was described in URI 50-400/02-11-05, Reliance on Manual Actions in Place of Required Physical Separation or Protection From a Fire.

- The corrective actions also created a manual operator action with excessive challenges such that there was not reasonable assurance that all non-licensed operators (NLOs) would be able to perform the action during a fire event. This finding was described in URI 50-400/02-11-06, Fire SSD Operator Actions With Excessive Challenges.

- In addition, the corrective actions created too many local manual operator actions for the new fire area for the one SSD NLO to perform. This finding was described in URI 50-400/02-11-07, Too Many Fire SSD Actions for Operators to Perform.

- Further, the corrective actions failed to provide the required emergency lighting for the new manual actions. This finding was described in URI 50-400/02-11-09, Failure to Provide Required Emergency Lighting for SSD Operator Actions.

Analysis: The inspectors and analysts evaluated the effects of the multiple examples of the overall finding of inadequate corrective action during a fire that could occur in the 1-A-ACP fire area of the RAB, using Phase 2 of the SDP. Based on that evaluation, the inspectors and analysts concluded that the overall finding had more than minor safety significance because it involved inadequate fire barriers for equipment that was

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relied upon for safe hot shutdown following a fire. The finding also had more than minor

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safety significance because it affected the availability and reliability objectives and the equipment performance attribute of the Mitigating Systems Cornerstone. In addition, it affected the Initiating Events Cornerstone in that it affected the objective of limiting the likelihood of occurrence of initiating events that challenge critical safety functions and also affected the design control attribute. The finding did not have more than very low safety significance (Green) because of the very low ignition sources in the fire area, manual suppression capability, and the power conversion system not being affected by a fire in this fire area. The Green significance determination was also confirmed by a walkdown of the fire area by two contractors.

Enforcement: Operating License Condition 2.F and the UFSAR, Section 9.5.1, FPP, includes quality assurance requirements for fire protection. The FPP states that a quality assurance program is being used to identify and rectify any possible deficiencies in design, construction, and operation of the fire protection systems. Operating License Condition 2.F requires that one of the redundant divisions be free of fire damage. Further, if both divisions were located in the same area, OLC 2.F requires that one of the divisions be physically protected from fire damage by one of three specified methods. Also, OLC.2.F requires that battery-backed emergency lights be provided in locations where operators are required to perform actions for SSD from a fire. In addition, Technical Specification 6.8.1, Procedures and Programs, requires procedures for implementing the FPP and for combating fires. The licensee's procedure for safe shutdown following a fire in the new ACP room fire area was AOP-36, Safe Shutdown Following a Fire, Rev. 24.

Contrary to the above requirements, the licensee's corrective actions for previous Violation 50-400/02-08-01 were inadequate because the actions failed to rectify deficiencies in design, construction, and operation related to SSD from a fire in the area of the ACP room. The licensee failed to protect various equipment either physically or procedurally from the effects of a fire where that equipment was relied on for SSD. The new ACP room effectively became part of the licensee's FPP when AOP-36 was revised (Revision 24) in November 2002 to include new operator actions for a fire in the new ACP room. Consequently, the conditions included in this violation were effectively in place for more than three days but less than thirty days when they were identified by the NRC in December 2002 during the triennial fire protection inspection. Because the identified examples of this inadequate corrective action are of very low safety significance and have been entered into the corrective action program (AR 80215), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 50-400/03-07-02; Inadequate Corrective Action for a Previous White Fire Protection Finding.

.2 (Closed) VIO 50-400/02-08-01, Failure to Implement and Maintain NRC Approved FPP SSD System Separation Requirements

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This Violation was closed with a new corrective action NCV opened as discussed in Section 1R05.1.b.(2).

.3 (Closed) URIs 50-400/02-11-01, -02, -03, -04, -05, -06, -07, -08, and -09

These URIs were resolved in two new NVCs as discussed in Sections 1R05.1.b.(1) and (2). Consequently, these URIs are closed.

4. OTHER ACTIVITIES

4OA3 Event Followup

(Closed) LER 50-400/02-04-00, Unanalyzed Condition Due to Inadequate Separation of Associated Circuits

This LER describes conditions that were previously identified by the NRC in IR 50-500/02-11 and that were evaluated and resolved in a new NCV in Section 1R05.1.b.(1) above. This LER was reviewed by the inspectors and no additional findings were identified. This LER is closed.

4OA6 Meetings, including Exit

The team presented the inspection results to Mr. R. Duncan and other members of his staff at the conclusion of the inspection on October 21, 2003. The licensee acknowledged the findings presented. Proprietary information is not included in this inspection report.

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SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

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F. Diya, Acting Manager, Engineering
R. Duncan, Director of Site Operations
M. Fletcher, Manager, Fire Protection Program
T. Morton, Manager, Support Services

NRC personnel

G. MacDonald, Senior Project Engineer, Division of Reactor Projects, Region II
R. Musser, Senior Resident Inspector, Shearon Harris
P. O'Bryan, Resident Inspector, Shearon Harris
C. Ogle, Chief, Engineering Branch 1, Division of Reactor Safety, Region II

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

50-400/03-07-01	NCV	Inadequate Implementation of the FPP for SSD [Section 1R05.1.b.(1)]
50-400/03-07-02	NCV	Inadequate Corrective Action for a Previous White Fire Protection Finding [Section 1R05.1.b.(2)]

Closed

50-400/02-08-01	VIO	Failure to Implement and Maintain NRC Approved FPP SSD System Separation Requirements (Section 1R05.2)
50-400/02-11-01	URI	Failure to Protect Charging System MOV 1CS-165, VCT Outlet to CSIPs, From Maloperation Due To a Fire (Section 1R05.3)

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50-400/02-11-02	URI	Failure to Protect Charging System MOVs 1CS-169, 1CS-214, 1CS-218, and 1CS-219 From Maloperation Due To a Fire (Section 1R05.3)
50-400/02-11-03	URI	Failure to Protect Charging System MOVs 1CS-166, 1CS-168, and 1CS-217 From Maloperation Due To a Fire (Section 1R05.3)
50-400/02-11-04	URI	Failure to Protect Component Cooling MOVs 1CC-251 and 1CC-208, CC for RCP Seals, From Maloperation Due To a Fire (Section 1R05.3)
50-400/02-11-05	URI	Reliance on Manual Actions in Place of Required Physical Separation or Protection From a Fire (Section 1R05.3)
50-400/02-11-06	URI	Fire SSD Operator Actions With Excessive Challenges (Section 1R05.3)
50-400/02-11-07	URI	Too Many Fire SSD Actions for Operators to Perform (Section 1R05.3)
50-400/02-11-08	URI	Using the Boric Acid Tank Without Level Indication (Section 1R05.3)
50-400/02-11-09	URI	Failure to Provide Required Emergency Lighting for SSD Operator Actions (Section 1R05.3)
50-400/02-04-00	LER	Unanalyzed Condition Due to Inadequate Separation of Associated Circuits (Section 4OA3)

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Attachment 1

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PROPRIETARY INFORMATION
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Attachment 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET SW SUITE 23T85
ATLANTA, GEORGIA 30303-8931

1

October 9, 2003

Carolina Power and Light Company
ATTN: Mr. J. S. Keenan
Vice President
Brunswick Steam Electric Plant
P. O. Box 10429
Southport, NC 28461

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT - NRC SAFETY SYSTEM DESIGN AND PERFORMANCE CAPABILITY INSPECTION - REPORT NOS. 05000325/2003008 and 05000324/2003008

Dear Mr. Keenan:

This refers to the safety system design and performance capability team inspection conducted on August 11-15 and August 25-29, 2003, at the Brunswick facility. The enclosed inspection report documents the inspection findings, which were discussed on August 29, 2003, with Mr. C. J. Gannon and other members of your staff.

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

Based on the results of this inspection, one finding of very low safety significance (Green) was identified. This issue was determined to involve a violation of NRC requirements. This finding has very low safety significance and has been entered into your corrective action program. However, the NRC is withholding the treatment of this issue as a non-cited violation as provided by Section VI.A.1 of the NRC's Enforcement Policy, pending our review of your corrective actions related to restoration of compliance. If you contest this finding, you should provide a response with the basis for your concern, within 30 days of the date of this inspection report to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Brunswick facility.

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.: 50-325, 50-324
License Nos.: DPR-71, DPR-62

Enclosure: NRC Inspection Report
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-325, 50-324
License Nos.:	DPR-71, DPR-62
Report Nos.:	05000325/2003008 and 05000324/2003008
Licensee:	Carolina Power and Light
Facility:	Brunswick Steam Electric Plant, Units 1 and 2
Location:	8470 River Road SE Southport, NC 28461
Dates:	August 11-15, 2003 August 25-29, 2003
Inspectors:	J. Moorman, Senior Reactor Inspector (Lead Inspector) N. Merriweather, Senior Reactor Inspector R. Schin, Senior Reactor Inspector (Week 1 only) M. Thomas, Senior Reactor Inspector M. Maymí, Reactor Inspector (Week 2 only) N. Staples, Reactor Inspector
Approved by:	Charles R. Ogle, Chief Engineering Branch 1 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000325/2003-008, 05000324/2003-008; 08/11-15/2003 and 08/25-29/2003; Brunswick Steam Electric Plant, Units 1 and 2; safety system design and performance capability.

This inspection was conducted by a team of inspectors from the Region II office. The team identified 1 Green unresolved item. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

Green. The team identified a violation of 10 CFR 50, Appendix B, Criterion III, Design Control requirements. The Technical Specification (TS) allowable value for the Condensate Storage Tank (CST) Level - Low function, for automatic high pressure coolant injection (HPCI) pump suction transfer to the suppression pool, was not adequately supported by design calculations. The calculations did not adequately address the potential for air entrainment in the HPCI process flow due to vortexing. This finding is in the licensee's corrective action program as Action Request 102456.

This finding is unresolved pending further NRC review of the requirements for the CST Level - Low function and of the corrective actions related to restoration of compliance with 10 CFR 50, Appendix B, Criterion III, Design Control requirements. The finding is greater than minor because it affects the design control attribute of the mitigating systems cornerstone objective. It is of very low safety significance (Green) because the finding is a design deficiency that will not result in loss of the HPCI function per GL 91-18 (Rev. 1) and the likelihood of having a low level in the CST that would challenge the CST level - low automatic HPCI suction transfer function is very low. In addition, alternate core cooling methods would normally be available, including reactor core isolation cooling (RCIC) as well as automatic depressurization system and low pressure coolant injection. (Section 1R21.11.b)

B. Licensee-Identified Violations

None

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events and Mitigating Systems

1R21 Safety System Design and Performance Capability (71111.21)

This team inspection reviewed selected components and operator actions that would be used to prevent or mitigate the consequences of a loss of direct current power event. Components in the high pressure coolant injection (HPCI), reactor core isolation cooling (RCIC), and 125/250 volt (v) direct current (dc) electrical systems were included. This inspection also covered supporting equipment, equipment which provides power to these components, and the associated instrumentation and controls. The loss of dc power event is a risk-significant event as determined by the licensee's probabilistic risk assessment.

.1 System Needs

.11 Process Medium

a. Inspection Scope

The team reviewed the licensee's installed configuration and calculations for water volume in the condensate storage tank (CST) and for net positive suction head for the HPCI pump. This included reviews of system drawings and walkdown 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 Technical Specification (TS) setpoint for the CST level instrumentation which initiates an automatic transfer 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.

b. Findings

Introduction: An unresolved item of very low safety significance (Green) was identified for inadequate design control of the HPCI suction source from the CST. The calculations which determined the CST low level setpoint for automatic HPCI system suction transfer from the CST to the suppression pool did not adequately account for air entrainment in the process flow due to vortexing. This finding involved a violation of NRC requirements. However, it is unresolved pending further NRC review of the requirements for the CST Level - Low function and corrective actions related to restoration of compliance.

Description: 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 air ingestion due to vortexing in containment heat removal systems. That guidance included limiting the Froude number (Fr) to less than 0.8 for BWR suppression pool suctions [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 experiments also showed that air entrainment increased dramatically when Fr reached 1.0. The BWR Owners' Group Emergency Procedure Guidelines included guidance on preventing vortexing in emergency core cooling system pump suctions from the suppression pool. This guidance included a vortex limit curve based on maintaining Fr less than 0.8.

All of the above references addressed suction pipes that extended into a tank/sump. A more recent research paper published in 2001 by ASME titled "Air Entrainment in a Partially Filled Horizontal Pump Suction Line" described tests on air entrainment. The tests were conducted at various flowrates, in a horizontal suction pipe that did not extend into the a tank; a configuration similar to the HPCI suction from the CST at Brunswick. The paper's conclusions about vortexing and air entrainment at high flow rates were similar to those of the previous references where a suction pipe extended into a tank.

Brunswick Units 1 and 2 TS Table 3.3.5.1-1 stated that the allowable value for the HPCI system automatic suction transfer from the CST to the suppression pool was a low CST level of ≥ 23 feet 4 inches above mean sea level. (NOTE: That value represented 3 feet 4 inches above the bottom of the CST.) Once initiated, the HPCI suction transfer involved first opening the suppression pool suction valves (E41-F041 and F042) and then closing the CST suction valve (E41-F004). The Updated Final Safety Analysis Report (UFSAR) stated that for each unit's CST:

> "...the HPCI and RCIC pumps take suction through a 16-inch line connected to the tank with a nozzle centerline 2 feet above the tank bottom. Level instruments will initiate an automatic transfer of the pumps' suction path to the suppression pool suction if level approaches this connection. For HPCI the setpoint is above the 3.3-foot TS limit and below the 3.5-foot calibration maximum allowed value. To allow time for the suction transfer to take place, this setpoint provides a margin of approximately 10,000 gallons in the tank after the setpoint is reached and before air will be entrained in the process flow."

The calculation of record that supported the TS allowable value was Calculation 0E41-1001, "High Pressure Coolant Injection System - Condensate Storage Tank Level - Low Uncertainty and Scaling Calculation [E41-LSL-N002(3) Loops]," Rev. 1, dated March 29, 1999. The team noted that Calculation 0E41-1001 stated that its objective was to determine the allowable value and setpoint for the CST low water level trip function for the HPCI system. However, the calculation did not include a hydraulic analysis to determine the allowable value. Instead, it relied on a design basis input from Engineering Service Request (ESR) 97-00026, Action Item 2, for the allowable value.

ESR 97-00026, Action Item 2, stated its objective: "... the analytical limit for the HPCI and RCIC CST low level transfer function is 23 feet 4 inches. Provide a basis for this analytical limit. The basis should address air voids ..." It also stated: "This ESR action item will show that using the TS limit as the analytical limit is acceptable." The ESR included Condition Report (CR) 97-02379 Task 2 (approved August, 27, 1997) as an attachment. The team noted that the ESR relied entirely on CR 97-02379 Task 2 for concluding that using the TS limit as the analytical limit was acceptable. However, the ESR also stated: "This CR review was not conducted as a design basis input with formal testing and design verification."

CR 97-02379 Task 2 stated that its objective was to determine if a vortexing problem existed in the CST when running the HPCI pump. Task 2 further stated that it was responding to an operating experience event where a nuclear plant had identified that they had failed to account for unusable volume in their CST due to vortexing concerns. It described a scale model test that had been performed by another nuclear plant to conclude that no vortexing would occur in their CST. However, the CR noted reasons why this test could not be relied upon as a design input. The CR also contained results from an informal test performed by the licensee. The CR concluded that, based on the results of the informal testing and engineering judgement, air ingestion may briefly occur during the transfer process; however, the air ingestion would be of such limited duration and such a small percentage that there was no concern for damage to the HPCI pumps. The team noted that the informal test used a small scale model without determination that the results would be applicable to the installed CST and HPCI suction, the test was performed without calibrated instruments, and the test was not independently verified. The team considered that the informal test was not suitable for use as an input to a design basis calculation.

Subsequently, action request (AR) 00005402 documented an engineering audit concern with relying on ESR 97-00026 as a design basis input to a calculation. ESR 01-00322 was then written to respond to AR 00005402. ESR 01-00322 stated that its purpose was to document the technical resolution of the CST intake vortex formation issue and to insert appropriate references into design documents. ESR 01-00322 included an extensive review of reference documents on vortexing. It included references to LERs and INPO Event Reports on vortexing issues at other nuclear plants; NUREG/CR-2772; and several research papers on vortexing. The team noted that ESR 01-00322 did not reference NRC Regulatory Guide 1.82.

ESR 01-00322 agreed with the conclusions of CR 97-02379 and ESR 97-00026 that the TS allowable value of 23 feet 4 inches was adequate. It concluded that the potential for a significant air ingestion event was of sufficiently low probability to be considered non-credible. The team noted that this conclusion was based primarily on the CR 97-02379 informal test and on a research paper by A. Daemi of the Water Research Center in Tehran, Iran, that had been presented to the American Society of Civil Engineers in

1998. The research paper tested the effect of an intake pipe protruding various distances into a reservoir and found that a pipe that did not protrude into the reservoir showed some vortexing but no air entrainment while a pipe that did protrude into the reservoir would have significant vortexing and air entrainment into the pipe. ESR 01-00322 considered that, since the NUREG/CR-2272 tests used a configuration where the suction pipe protruded into the tank and the licensee's HPCI suction pipe did not protrude into the CST, the NUREG/CR-2272 conclusions were not applicable to the Brunswick design. The NRC team noted that the research paper by A. Daemi was significantly flawed for applicability to Brunswick in that it did not state what flowrates were used in its tests and apparently used gravity flow. Regulatory Guide 1.82 and NUREG/CR-2272 indicate that flow velocity is one of the most important factors in vortex formation. A suction pipe that would have little or no vortexing at low flow velocities (e.g., gravity flow) could have significant vortexing at higher flow velocities (e.g., a HPCI pump at 4300 gpm). The team considered that both sources of information on which the conclusions of ESR 01-00322 were based were not suitable for use as inputs to safety-related design calculation 0E41-1001.

The HPCI pump was designed to automatically start and establish a flowrate of 4300 gpm. Licensee procedures did not contain guidance to reduce that flowrate when the CST level approached the low level switchover setpoint. Using the NUREG/CR-2272 methodology, the team calculated that, at a HPCI pump flowrate of 4300 gpm, an Fr of 0.8 would be reached at a CST level of 5.0 feet and an Fr of 1.0 would be reached at a CST level of 3.9 feet. Considering the automatic suction transfer actuation setpoint and the valve stroke times, the HPCI pump suction pipe could be exposed to a suction Fr in excess of 0.8 (some air entrainment) for about 8.9 minutes and over 1.0 (over 2% air entrainment) for about 5.0 minutes. Calculations that used the 2001 ASME research paper equations provided different results: air entrainment in the process flow would start at a tank level of 3.2 feet and would exceed 2% at tank levels below 3.0 feet. This would represent a HPCI pump suction pipe exposure to some air entrainment in the process flow for about 1.8 minutes and to over 2% air entrainment for about 1.1 minutes. The team concluded that the plant design was not consistent with the UFSAR in that the TS allowable value for the HPCI automatic suction transfer would not prevent air from becoming entrained in the HPCI process flow.

During this inspection, team and licensee measurements of the installed CST configuration revealed non-conservative errors of about 1.5 inches in the actual heights of the Units 1 and 2 CST level switches above the HPCI suction pipes. These would result in additional non-conservative errors in the HPCI automatic suction transfer setpoints.

The licensee entered this issue into their corrective action program as AR 102456. This AR included an operability determination and planned corrective actions that were reviewed by the team. The operability determination concluded that the CST Level - Low instrument was operable with the existing TS allowable value and related setpoint and no compensatory measures were needed. This conclusion was based on the following: 1) HPCI operation during design or licensing basis events would not challenge the CST Level - Low instrument; and 2) Operator actions consistent with plant procedures would not result in 4300 gpm HPCI flow for the full duration of the suction transfer. The operability determination did not include an analysis which assured that the instrument's allowable value was adequate to prevent significant air entrainment during the full duration of a CST Level - Low setpoint initiated suction transfer while the HPCI pump was operating at its maximum flowrate of 4300 gpm. However, the team's interpretation of licensing basis documents indicated that the CST Level - Low function was required to be able to protect the HPCI pump from damage from any suction hazard that could occur. This included air entrainment in the process flow due to vortexing that would result if the CST level became low while the HPCI pump was operating at about 4300 gpm, even if this could only occur outside of a design basis event.

The licensee's corrective actions for this issue were in AR 102456. This AR included only two planned corrective actions. The first corrective action was: "Issue a UFSAR change package to correct the description of HPCI air entrainment potential during suction swap." This was described in more detail in the AR under Section 3, Inappropriate Acts, item 4: "Error 4 was a simple text error by BNP engineering where the concept was understood (no significant air at the pump) but was not translated into specific detailed words." The second corrective action was: "Issue an evaluation to update the HPCI CST level switch design basis information to reflect the evaluation provided in the operability review portion of this AR." The operability determination portion of the AR concluded that the CST Level - Low automatic HPCI suction transfer function would not be challenged during design basis events and consequently the TS allowable value was adequate.

The documented corrective actions in AR 102456 did not appear to be sufficiently comprehensive to restore compliance with 10 CFR 50, Appendix B, Criterion III, Design Control. The licensee's planned corrective actions did not specifically include revising the design calculation, 0E41-1001. In addition, they did not include assuring that the CST Level - Low suction transfer function will protect the HPCI pump if it is operating at its maximum flowrate during the transfer. The planned corrective actions identified in the AR did not include obtaining a certification from the pump vendor that the pump can withstand a certain amount of air in the process flow for a certain amount of time without pump damage. [This was subsequently done by the licensee.] The planned corrective actions identified in the AR also did not include submitting a license amendment request to the NRC to revise the TS allowable value, remove the CST Level - Low function from TS, or add an operator action to throttle HPCI pump flow at low CST levels so that the existing setpoint will be able to protect the pump. This issue will remain unresolved pending further NRC review of the design basis and operability requirements for the CST Level - Low suction transfer function. Specifically, the NRC will review whether the CST Level - Low function is required to be able to protect the HPCI pump from damage only during design basis events; or if it is required to be able to protect the HPCI pump from damage due to air entrainment if the level is the CST becomes low with the HPCI pump operating at a flowrate of about 4300 gpm, even if this could only occur outside of a design basis event.

Analysis: Design Calculation 0E41-1001, for the CST Level - Low setpoint and TS allowable value was inadequate. The finding is greater than minor because it affects the design control attribute of the mitigating systems cornerstone objective. It is of very low safety significance (Green) because the finding is a design deficiency that will not result in loss of the HPCI function per GL 91-18 (Rev. 1) and the likelihood of having a low level in the CST that would challenge the CST Level - Low automatic HPCI suction transfer function is very low. In addition, alternate core cooling methods would normally be available, including RCIC as well as automatic depressurization system and low pressure coolant injection.

Enforcement: 10 CFR 50, Appendix B, Criterion III, Design Control, requires in part, that design control measures shall include provisions to assure that appropriate quality standards are specified and included in design documents. Contrary to the above requirements, the NRC identified during this inspection that, from 1999 to August 2003, licensee Calculation 0E41-1001 and associated design documents did not adequately consider air entrainment in the HPCI pump process flow due to vortexing in the CST for the current TS value for the CST Level - Low setpoint for automatic transfer of the HPCI pump suction from the CST to the suppression pool. This finding was entered into the licensee's corrective action program as Action Request 102456 and is unresolved pending further NRC review of the requirements for the CST Level - Low function and of the licensee's corrective actions related to restoration of compliance with Criterion III of 10 CFR 50, Appendix B. This finding is identified as URI 05000325, 324/2003008-01, Failure to Adequately Consider Vortexing in the Calculation for CST Level for Automatic Transfer of the HPCI Pump Suction.

.12 Energy Sources

a. Inspection Scope

The team reviewed appropriate test and design documents to verify that the 125/250 vdc power source for HPCI system valves and controls would be available and adequate in accordance with design basis documents. Specifically, the team reviewed the 125/250 vdc battery load study, 125 vdc battery charger sizing calculation, and 125/250 vdc system voltage drop study, and battery surveillance test results, to verify that the dc batteries and chargers had adequate capacity for the loading conditions which would be encountered during various operating scenarios. The team reviewed a sample of HPCI motor operated valves (MOVs) to verify the adequacy of available motor output torque, stroke times, thermal overload heater sizing, and valve performance at reduced voltages. The team also reviewed portions of a voltage study to verify adequacy of voltage for HPCI solenoid valves 1-E41-F025 and -F026 under worst case voltage conditions. A list of related documents reviewed are included in the attachment.

The team reviewed design basis descriptions and drawings and walked down the HPCI and RCIC systems to verify that a steam supply would be available for pump operation during a loss of station dc power event. This included review of the steam supply drain systems and review of a recent modification to the HPCI steam supply drain system. The team reviewed the HPCI steam supply drain pot flow orifice inspections; the drain pot level switch logic and calibration records, and the drain pot drain line isolation valves modification to verify that the HPCI steam supply would be available if needed. The team reviewed functional valve testing for the HPCI and RCIC turbine exhaust vacuum breaker check valves to verify adequacy of acceptance criteria and to verify that vacuum breaker functionality was being maintained.

b. Findings

No findings of significance were identified.

.13 Instrumentation and Controls

a. Inspection Scope

The team reviewed electrical elementary and logic diagrams depicting the HPCI pump start and stop logic, permissives, and interlocks to ensure that they were consistent with the system operational requirements described in the UFSAR. The team reviewed the HPCI auto-actuation and isolation functional surveillance procedures and completed test records to verify that the control system would be functional and provide desired control during accident and event conditions in accordance with design. The team reviewed the calibration test records for the CST low water level instrument channels to verify that the instruments were calibrated in accordance with setpoint documents. The team also reviewed the records demonstrating the calibration and functional testing of the HPCI suppression pool high level instrument channels to determine the operability of the high level interlock functions of HPCI.

b. Findings

No findings of significance were identified.

.14 Operator Actions

a. Inspection Scope

The team assessed the plant and the operators' response to a Unit 1 initiating event involving a loss of station battery 1B-2. The team focused on the installed equipment and operator actions that could initiate the event or would be used to mitigate the event. The team reviewed portions of emergency operating procedures (EOPs), abnormal operating procedures (AOPs), annunciator panel procedures (APPs), and operating procedures (OPs) to verify that the operators could perform the necessary actions to respond to a loss of dc power event. The team also observed simulation of a loss of dc power event on the plant simulator and walked down portions of Procedure 0AOP-39, "Loss of DC Power." The simulator observations and procedure reviews focused on plant response and on verifying that operators had adequate instrumentation and procedures to respond to the event. The team reviewed operator training records (lesson plans, completed job performance measures, etc.) to verify that operators had received training related to a loss of dc power event.

b. Findings

No findings of significance were identified.

.15 Heat Removal

a. Inspection Scope

The team reviewed historical temperature data for the Unit 2 battery rooms to verify that the minimum and maximum room temperatures were within the allowable temperature limits specified for the batteries.

The team reviewed heat load and heat removal calculations for the HPCI and RCIC rooms. The team also reviewed the calculated peak temperature and pressure responses during high energy line break and loss of coolant accidents for these rooms. The team reviewed service water temperature and flow requirement calculations for the HPCI and RCIC rooms and fan coolers. These reviews were conducted to verify the adequacy of design for the room coolers, and to verify that heat will be adequately removed during a loss of dc power event.

The team also reviewed HPCI and RCIC room cooler thermostat calibrations, inspection and cleaning records, and corrective maintenance history to verify room coolers were properly maintained and would be available if called upon.

b. Findings

No findings of significance were identified.

.2 System Condition and Capability

.21 Installed Configuration

a. Inspection Scope

The team visually inspected the 125/250 vdc batteries and battery chargers, dc distribution panels, dc switchgear, and dc ground detection systems in both units to verify that the dc system was in good material condition with no alarms or abnormal conditions present and to verify that alignments were consistent with the actions needed to mitigate a loss of dc power event. The batteries were inspected for signs of degradation such as corrosion, cell discoloration, plate buckling, grid cracks, and excessive plate growth.

The team walked down the HPCI and RCIC systems and the CST to verify that the installed configuration was consistent with design basis information and would support system function during a loss of dc power event.

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 dc power event. During this walkdown, the team compared valve positions with those specified in the HPCI system operating procedure lineup, and observed the material condition of the plant to verify that it would be adequate to support operator actions to mitigate a loss of dc power event. This also included reviewing completed surveillance tests which verified selected breaker positions and alignments.

b. Findings

No findings of significance were identified.

.22 Design Calculations

a. Inspection Scope

The team reviewed the thermal overload sizing calculations for a sample of Unit 1 HPCI MOVs to verify adequacy of the installed overload relay heaters. The team also reviewed calculations that assessed the stroke times and motor torque produced at reduced voltage to verify that they would exceed or meet minimum specified requirements. The valves and calculations reviewed are listed in the attachment.

The team reviewed design basis documents, probabilistic risk assessment system notebooks, UFSAR, selected piping and instrumentation diagrams, selected TSs, system reviews, ARs, and the corrective maintenance history for HPCI and RCIC systems to assess the implementation and maintenance of the HPCI and RCIC design basis.

b. Findings

No findings of significance were identified.

.23 Testing and Inspection

a. Inspection Scope

The team reviewed the 125/250 vdc battery surveillance test records, including performance and service test results, to verify that the batteries were capable of meeting design basis load requirements.

The team reviewed functional and valve operability testing (stroke times), and corrective maintenance records for HPCI and RCIC selected valves, including the minimum flow bypass valves, and steam admission valve. This review was conducted to verify the availability of the selected valves, adequacy of surveillance testing acceptance criteria, and monitoring of selected valves for degradation.

The team reviewed HPCI and RCIC system operability tests to verify the adequacy of acceptance criteria, pump performance under accident conditions, and monitoring of system components for degradation.

b. Findings

No findings of significance were identified.

.3 Selected Components

.31 Component Degradation

a. Inspection Scope

The team reviewed in-service trending data for selected components, including the HPCI and RCIC pumps, to verify that the components were continuing to perform within the limits specified by the test.

The team reviewed the maintenance history of the 125/250 vdc batteries, 125 vdc battery chargers, and selected 4160 v alternating current (ac) and 480 vac breakers 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 preventive maintenance performed on selected 4160 vac and 480 vac breakers to verify that preventive maintenance was being performed in accordance with maintenance procedures and vendor recommendations. The specific work orders and other related documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.32 Equipment/Environmental Qualification

a. Inspection Scope

The team conducted in-plant walkdowns to verify that the observable portion of selected mechanical components and electrical connections to those components were suitable for the environment expected under all conditions, including high energy line breaks.

b. Findings

No findings of significance were identified.

.33 Equipment Protection

a. Inspection Scope

The team conducted in-plant walkdowns to verify that there was no observable damage to installations designed to protect selected components from potential effects of high winds, flooding, and high or low outdoor temperatures.

The team walked down the HPCI and RCIC systems and the CST to verify that they were adequately protected against external events and a high energy line break.

b. Findings

No findings of significance were identified.

.34 Operating Experience

a. Inspection Scope

The team reviewed the licensee's dispositions of operating experience reports applicable to the loss of dc power event to verify that applicable insights from those reports had been applied to the appropriate components.

b. Findings

No findings of significance were identified.

.4 Identification and Resolution of Problems

a. Inspection Scope

The team reviewed corrective maintenance work orders on batteries, battery chargers, and ac breakers to evaluate failure trends. The team also reviewed Action Requests involving battery problems, battery charger problems, and charger output breaker problems to verify that appropriate corrective action had been taken to resolve the problem. The specific Action Requests reviewed are listed in the attachment. The team reviewed selected system health reports, maintenance records, surveillance test records, calibration test records, and action requests to verify that design problems were identified and entered into the corrective action program.

b. Findings

No findings of significance were identified.

4. Other Activities

4OA6 Meetings, Including Exit

The lead inspector presented the inspection results to Mr. C. J. Gannon, and other members of the licensee staff, at an exit meeting on August 29, 2003. The inspectors confirmed that proprietary information was not provided or examined during this inspection.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

L. Beller, Supervisor, Licensing
E. Browne, Engineer, Probabilistic Safety Assessment
B. Cowan, Engineer
C. Elberfeld, Lead Engineer
P. Flados, HPCI System Engineer
N. Gannon, Director, Site Operations
M. Grantham, Design
C. Hester, Operations Support
D. Hinds, Manager, Engineering
G. Johnson, NAS Supervisor
W. Leonard, Engineer
T. Mascareno, Operations Support
J. Parchman, Shift Technical Advisor, Operations
C. Schacher, Engineer
B. Stackhouse, Systems
H. Wall, Manager, Maintenance
K. Ward, Technical Services

NRC (attended exit meeting)

E. DiPaolo, Senior Resident Inspector
J. Austin, Resident Inspector

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000325,324/2003008-01 URI Failure to Adequately Consider Vortexing in the Calculation for CST Level for Automatic Transfer of the HPCI Pump Suction (Section 1R21.11. b)

LIST OF DOCUMENTS REVIEWED

Procedures

0AI-115, 125/250 VDC System Ground Correction Guidelines, Rev. 6
0AOP-36.1, Loss of Any 4160V Buses or 480V E-Buses, Rev. 25
0AOP-39.0, Loss of DC Power, Rev. 16
0OI-01.02, Shift Routines and Operating Practices, Rev. 31
0OI-50, 125/250 VDC Electrical Load List, Rev. 25
0OP-50.1, Diesel Generator Emergency Power System Operating Procedure, Rev. 55
0PM-ACU500, Inspection and Cleaning of the RHR/Core Spray Room Aerofin Cooler Air Filters and Coolers, Rev. 7
1APP-A-05, Annunciator Procedure for Panel A-05, Rev. 46
1APP-UA-23, Annunciator Procedure for Panel UA-23, Rev. 45
1EOP-01-RSP, Reactor Scram Procedure, Rev. 8
1OP-19, High Pressure Coolant Injection System Operating Procedure, Rev. 58
1OP-50, Plant Electrical System Operating Procedure, Rev. 64
1OP-51, DC Electrical System Operating Procedure, Rev. 40
2APP-A-01, Annunciator Procedure for Panel A-01, Rev. 44
OPIC-TMR002, Calibration of Agastat 7020 Series Time Delay Off Relays, Rev. 18
OPM-BKR001, ITE 4KV-line Breaker and compartment checkout, Rev 27
OPM-BKR002A, ITE K-line Circuit Breakers, Rev 31
OPM-TRB518, HPCI & RCIC Steam Inlet Drain Pot Flow Orifices Inspection, Rev. 3

Drawings

1-FP-60085, High Pressure Coolant Injection System Unit 1, Rev. J
Contract No. 71-2162, Dwg. No. 1, General Plan for Condensate Storage Tanks by Brown & Root, Inc; Rev. C
D-02523, High Pressure Coolant Injection System Unit 2, Sh. 1 & 2, Rev. 52 & 45
D-02529, Reactor Core Isolation Cooling System Unit 2, Sh. 1 & 2, Rev. 52 & 36
D-25023, Sheet 2, Unit 1 High Pressure Coolant Injection System Piping Diagram, Rev. 45
D-25023, Sheet 1, Unit 1 High Pressure Coolant Injection System Piping Diagram, Rev. 54
F-03044, Units 1 & 2 480 Volt System Key One Line Diagram, Rev. 18
LL-7044, Instrument Installation Details Units 1 & 2, Sh. 15, Rev. 10

Calculations

0E41-1001; High Pressure Coolant Injection System - Condensate Storage Tank Level - Low Uncertainty and Scaling Calculation (E41-LSL-N002(3) Loops), Rev. 1, dated March 29, 1999
9527-8-E41-06-F; NPSH Requirements - HPCI and RCIC; dated March 26, 1987
BNP-E-6.033, AC/DC MOV Thermal Overload Sizing Calculations, Rev. 3
BNP-E-6.062, 125/250 Volt DC System Voltage Drop Study, Rev. 3
BNP-E-6.074, 125/250 Volt DC Battery Load Study, Rev. 2
BNP-E-6.079, 125 Volt DC Battery Charger Sizing Calculation, Revision
BNP-E-6.109, Unit 1 Stroke and Motor Torque Calculations for 250VDC Safety-Related MOVs, Rev. 5
BNP-E-8.013, Motor Torque Analysis for AC MOVs, Rev. 4

BNP-EQ-4.001, Temperature Response in RHR and HPCI Rooms Following LOCA with Reduced
BNP-MECH-E41-F002, Mechanical Analysis Report to Verify Minimum Torque Availability, Rev. 3
BNP-MECH-RBER-001, Reactor Building Environmental Report, Rev. 0A
HVAC Flow Rates, Rev. 0
M-89-0021; HPCI/RCIC NPSH with Suction from the CST; Rev. 0, dated November 27, 1989
PCN-G0050A, RHR Room Cooler Allowable Service Water Inlet Temperature, Rev. 2

Design Basis Documents

DBD-19, High Pressure Coolant Injection System, Rev. 11
DBD-51, DC Electrical System, Rev. 5

Engineering Service Requests

ESR 97-0026; Provide a Basis for the Analytical Limit for the HPCI and RCIC CST Low Level Transfer Function; dated November 24, 1997
ESR 98-00067; HPCI/RCIC Reserve Capacity in CST; Rev. 1, dated February 17, 1998
ESR 99-00404; HPCI/RCIC Drain Pot Piping Boundary Changes; dated February 25, 2000
ESR 01-00322; Document the Technical Resolution of the CST Intake Vortex Formation Issue; dated September 25, 2001
ESR 99-00405, HPCI Design Conversion To Fail Open for E-41-F028/29, Rev. 0

Updated Final Safety Analysis Report

UFSAR Section 5.4.6, Reactor Core Isolation Cooling System
UFSAR Section 6.3, Identification of Safety Related Systems - Emergency Core Cooling Systems
UFSAR Section 7.1.1.2, Emergency Core Cooling Systems
UFSAR Section 8.3.2, DC Power Systems
UFSAR Section 9.2.6, Condensate Storage Facilities

Improved Technical Specifications

Section 3.5.1, ECCS - Operating
Section 3.5.3, RCIC System
Section 3.8.4, DC Sources - Operating
Section 3.8.6, Battery Cell Parameters
Section 3.8.7, Electrical Distribution Systems - Operating
TS Bases Section 3.5; Emergency Core Cooling Systems and Reactor Core Isolation Cooling System

List of Valves Inspected
1-E41-F001, HPCI Steam Supply Valve
1-E41-F006, HPCI Main Pump Discharge Valve
1-E41-F007, HPCI Main Pump Discharge Valve
1-E41-F008, HPCI Test Bypass to CST Valve

1-E41-F011, HPCI Redundant Shutoff to CST Valve
1-E41-F012, HPCI Test Line Miniflow Valve
1-E41-F041, HPCI Suppression Pool Suction Valve
1-E41-F042, HPCI Pump Suction Valve

Completed Maintenance and Tests

0PT-09.2, HPCI System Operability Test, completed 06/27/03, 04/03/03, 01/10/03, 08/20/03, 05/29/03, 04/04/03
0PT-20.10, Testing of Valves E41-F076, E41-F077, E51-F063, E51-F064, completed 04/24/02, 03/08/02, 03/10/03, 04/22/02
0PT-10.11, RCIC System Operability Test, completed 06/06/03, 03/14/03, 12/20/02, 07/31/03, 05/08/03, 04/03/03
0PT-09.3, HPCI System - 165 Psig Flow Test, completed 04/20/03, 03/26/01, 03/29/02, 03/23/00
0PT-09.7, HPCI System Valve Operability Test, completed 07/25/03, 05/02/03, 02/07/03, 05/01/03, 04/01/03
0PT-10.1.8, RCIC System Valve Operability Test, completed 07/04/03, 04/10/03, 07/03/03, 04/09/030PT-10.1.3, RCIC System Operability Test - Flow Rates at 150 Psig, completed 03/18/00, 03/29/02, 03/23/01, 04/02/03

Completed Work Orders (WOs) and Work Requests (WRs)

WO 49443-01, HPCI Turbine Restricting Orifices Inspection, completed 03/13/01
WO 49442-01, RCIC Turbine Restricting Orifices Inspection, completed 03/15/01
WO 45798-01, HPCI Turbine Supply Steam Drain Pot Hi Level Switch Calibration (Unit 2), completed 02/06/01
WO 192543-01, HPCI Steam Supply Valve 2-E41-F001 Repairs due to Leakage Past the Seat, completed 03/31/03
WO 45817-01, HPCI Turbine Supply Steam Drain Pot Hi Level Switch Calibration (Unit 1), completed 11/25/01
WO 46107-01, Calibration of RHR Room Cooler Thermostats, completed 11/09/00
WO 50172-01, Inspection & Cleaning of the RHR Room Cooler, completed 03/05/02
WO 50171-01, Inspection & Cleaning of the RHR Room Cooler, completed 03/05/02
WR AFQO 001, HPCI Turbine Supply Steam Drain Pot Hi Level Switch Calibration (Unit 2), completed 06/07/96
WR AITI 001, HPCI Turbine Supply Steam Drain Pot Hi Level Switch Calibration (Unit 1), completed 08/03/95
WR ABPD 003, Calibration of RHR Room Cooler Thermostats, completed 09/13/00
WR ABPD 002, Calibration of RHR Room Cooler Thermostats, completed 06/25/97
WR AGEB 002, Calibration of RHR Room Cooler Thermostats, completed 08/21/97
WR AIWK 004, Inspection & Cleaning of the RHR Room Cooler, completed 03/09/02
WR/JO ANRR001, 1A-1 Batteries, 125 VDC, Performance Capacity Test
WR/JO ANTK001, 1A-2 Batteries, 125 VDC, Performance Capacity Test
WR/JO ANSN001, 1B-1 Batteries, 125 VDC, Performance Capacity Test
WR/JO ANST001, 1B-2 Batteries, 125 VDC, Performance Capacity Test
WO 0004546501, 2B-1 Batteries, 125 VDC, Performance Capacity Test
WO 0004546401, 2B-2 Batteries, 125 VDC, Performance Capacity Test

WO 0004546301, 2A-1 Batteries, 125 VDC, Performance Capacity Test
WO 0004546601, 2A-2 Batteries, 125 VDC, Performance Capacity Test
WO 0004635001, 1A-2 Batteries, 125 VDC, Service Capacity Test
WO 0004635101, 1A-1 Batteries, 125 VDC, Service Capacity Test
WO 0004634901, 1B-1 Batteries, 125 VDC, Service Capacity Test
WO 0004634801, 1B-2 Batteries, 125 VDC, Service Capacity Test
WO 0017812801, 2B-2 Batteries, 125 VDC, 2B-2 Service Capacity Test
WO 0017569601, 2B-1 Batteries, 125 VDC, 2B-1 Service Capacity Test
WO 0017400501, 2A-1 Batteries, 125 VDC, 2A-1 Service Capacity Test
WO 0017414101, 2A-2 Batteries, 125 VDC, 2A-2 Service Capacity Test
WO 0040723401, OMST-BATT11W, 125 VDC, Weekly Test
WO 0040495901, OMST-BATT11W, 125 VDC, Weekly Test
WO 0040496001, OMST-BATT11W, 125 VDC, Weekly Test
WO 0040777701, OMST-BATT11W, 125 VDC, Weekly Test
WO 0037714901, 1B-1 & 1B-2 OMST-BATT11Q Quarterly
WO 0031256501, 1B-1 & 1B-2 OMST-BATT11Q Quarterly
WO 0030900101, 1B-1 & 1B-2 OMST-BATT11Q Quarterly
WO 0028260501, 1B-1 & 1B-2 OMST-BATT11Q Quarterly
WO 0038119301, 1A-1 & 1A-2 OMST-BATT11Q Quarterly
WO 0031639601, 1A-1 & 1A-2 OMST-BATT11Q Quarterly
WO 0031256401, 1A-1 & 1A-2 OMST-BATT11Q Quarterly
WO 0028260601, 1A-1 & 1A-2 OMST-BATT11Q Quarterly
WO 0030391401, 2A-1 & 2A-2 OMST-BATT11Q Quarterly
WO 0030391501, 2B-1 & 2B-2 OMST-BATT11Q Quarterly
WO 0031256201, 2A-1 & 2A-2 OMST-BATT11Q Quarterly
WO 0031256301, 2A-1 & 2A-2 OMST-BATT11Q Quarterly
WO 0031256601, 2B-1 & 2B-2 OMST-BATT11Q Quarterly
WO 0031256701, 2B-1 & 2B-2 OMST-BATT11Q Quarterly
WO 0004680801, HPCI Auto-Actuation and Isolation Logic System Functional Test
WO 0017956801, HPCI Auto-Actuation and Isolation Logic System Functional Test
WO 0039711701, 1MST-HPCI27Q and RCIC CST Low Water Level Instrument Calibration
WO 0031316101, 1MST-HPCI27Q and RCIC CST Low Water Level Instrument Calibration
WO 0039317801, 2MST-HPCI27Q and RCIC CST Low Water Level Instrument Calibration
WO 0031323101, 2MST-HPCI27Q and RCIC CST Low Water Level Instrument Calibration
WO 0038677201, HPCI Suppression Pool High Level Instrument Channel Calibration
WO 0031264601, HPCI Suppression Pool High Level Instrument Channel Calibration
WO 0038677301, HPCI Suppression Pool High Level Instrument Channel Calibration
WO 0004589001, Calibrate 1-E41-FSHL-N006 in accordance with OPIC-DP-S001
WO 0007165101, Replace HPCI pump discharge line flow switch
WO 0043163601, Perform single cell charging on 1-1A-2 Cell #43 IAW OSPP-BAT010
WO 0043161301, Perform single cell charging on 1-1B-1 Cell #13 IAW OSPP-BAT010
WO 0042888401, 1-1B-1 125 VDC Battery Cell # 13 has a low voltage reading
WO 0041657401, Perform single cell charging on 1A-2 Battery Cell # 1
WO 0037821401, 1B-2 Battery Cell # 53 has a cell voltage of 2.124, minimum voltage is 2.13
WO 0033286001, 1-1B-2 Battery corrosion found on positive terminal of battery cell # 52
WO 0033285401, 1-1A-1 Battery corrosion found
WO 0033285301, 1-1A-2-125VDC-BAT, Replace Cell # 7 on Battery 1A-2
WO 0016351401, Equalize 1-1B-2-125VDC-BAT IAW OPM-BAT004

WO 0014092401, 1-1B-2 Cell # 1 needs to be replaced due to low specific gravity reading
WO 0006930901, Using ESR 00-00345 and WO Task Instructions, Replace Cell # 54 in 1-1B-125VDC-BAT while batteries remain on line
WO WR/JO 99-ADIK1, Troubleshoot and assist operations in ground hunting for 1B Battery Bus IAW OAI-115 and 1OP-51
WO 0043131301, 1-1A-2-125VDC-CHRGR investigate breaker trip/charger voltage card replacement
WO WR/JO 99-AFEC1, Replace float/equalize toggle switch on 1-1A-1-125VDC-CHRGR
WO WR/JO 99-AFED1, Replace float/equalize toggle switch on 1-1A-2-125VDC-CHRGR
WO WR/JO 99-AFEE1, Replace float/equalize toggle switch on 1-1B-1-125VDC-CHRGR
WO WR/JO 99-AFEE2, Place 1-1B-1-125VDC-BAT on equalize
WO WR/JO 99-AGKA1, Investigate problem with 1-1B-2-125VDC-CHRGR
WO WR/JO 99-AGKA2, Troubleshoot ground on 1-1B-2 Battery Charger during Unit 1 outage
WO WR/JO 99-AFEF1, Replace float/equalize toggle switch on 1-1B-2-125VDC-CHRGR
WO WR/JO 98-ACNW1, Troubleshoot and Repair 1-1B-2-125VDC-CHRGR
WO 0033286301, Perform OMST-BATT11Q to remove corrosion from battery terminals
WO 0033286201, Perform OMST-BATT11Q to remove corrosion
WO 0027849301, 2-2A-1-125VDC-BAT, Perform DLRO measurements
WO 0027849201, 2-2B-1-125VDC-BAT, Perform DLRO measurements
WO 0016331601, 2-2B-1-125VDC-CHRGR has no output voltage please investigate and repair
WO 0013345101, The corrected specific gravity was less than the required 1.205 tolerance
WO WR/JO 99-ADML1, Place 125 VDC Battery Banks 2A-1, 2A-2, 2B-1, 2B-2 on equalize
WO WR/JO 00-ADJS1, Replace Cell # 27 in 2-2A-2-125VDC-BAT
WO WR/JO 00-ADEE1, Clean off electrolyte on cell #27 of 2-2A-2 Battery
WO WR/JO 99-AAGJ1, 2-2A-2-125VDC-BAT individual cell voltage out of tolerance
WO WR/JO 00-AARJ1, Troubleshoot 2-2B battery bus ground
WO WR/JO 99-ACRS1, Replace float/equalize toggle switch on 2-2A-2-125VDC-CHRGR
WO WR/JO 99-ACSW1, Replace float/equalize toggle switch on 2-2A-1-125VDC-CHRGR
WO 0011166201, Replace float/equalize toggle switch on 2-2B-1-125VDC-CHRGR
WO 0017170101, Specific gravity on Cell #56 of battery 1B-2 out of tolerance
WO WR/JO 99-AAGE1, 1-1B-2-125VDC-BAT Cell #37 voltage low
WR/JO ASLE001,1-E6-AV4-52, 5175 480 VAC Distribution System, Substation Breaker PM
WR/JO ADUE001,1-E5-AU9-52, 5175 480 VAC Distribution System, Substation Breaker PM
WR/JO ADKC001,1-E6-AX1-52, 5175 480 VAC Distribution System, Substation Breaker PM
WR/JO 99-ACPT1,2-2CB-C56, 5175 480 VAC Distribution System, Substation Breaker Maintenance
WR/JO 00-ABHD2,1-1CA-C05, 5175 480 VAC Distribution System, Substation Breaker Maintenance
WR/JO 00-ABDH1,1-1CA-C05, 5175 480 VAC Distribution System, Substation Breaker Maintenance
WR/JO ACDU001, 2-2A-GKO-72, 5240 125 VDC Battery Charger System, Circuit Breaker Functional Test
WR/JO ACDX001, 2-2A-GK3-72, 5240 125 VDC Battery Charger System, Circuit Breaker Functional Test
WR/JO AAKO001, 2-2CB-C56-52, 5240 125 VDC Battery Charger System, Circuit Breaker Maintenance
WO 0005034401, PM on 1-E2-AH1
WO 0017871402, In-situ Test of Mag Latch for 1-E6-AV4-52

WO 0030223001, Overload Relay Setting Change
WO 0017871002, In-situ Test on 1-E6-AV4-52
WO 0029973501, Circuit Breaker Tie Between Unit Substation E5&E6
WO 0017868201, In-situ Test of Mag Latch of E5-E6 Tie Breaker
WO 0005033201, PM on 1-E2-AH1
WO 0012789501, Breaker Operator Replacement
WO 0005030701, PM on Breaker 1-1B-GM1-72
WO 0005009301, PM on Breaker 1-1B-GM4-72
WO 0029610701, PM on Breaker 2-2B-GM1-72
WO 0029609301, PM on Breaker 2-2B-GM4-72
WO 0013432712, Test/Replace Breaker 2B-1-125VDC-Charger AC CKT

Completed Surveillance Procedures, Preventive Maintenance (PM), and Test Records

0PT-12.6, Breaker Alignment Surveillance, Rev. 42, Completed 8/2/03, 8/9/03, 8/16/03, 8/23/03

Action Requests (ARs)

087358, Deficiencies related with valve 2-E41-F001
CR 97-02379; Determine if Vortexing Problem Exists in the CST When Running the HPCI Pump; dated August 27, 1997.
AR 00005402; Vortexing in CST Needs More Formal Analysis than CR 97-02379; dated December 30, 1998.
AR 00098654, 125 VDC 1A-2 Battery Charger Main Supply Breaker Trip
AR 00047078, 1B-2 Cell # 56 Failed Specific Gravity
AR 00071076, Positive Plate Discoloration and Expansion
AR 00071079, 1B-2 Battery cells have positive plate discoloration and expansion
AR 00058078, Battery 1A-2 has low voltage cells
AR 00053109, Visual signs of degradation on 2B-1 battery
AR 00083997, 2A-1 Battery Cell #31 cracked cell top
AR 00085750, 1B-2 Battery Cell #53 has a low voltage
AR 00044684, 1B-2 Batteries are A(1) under new Maintenance Rule criteria
AR 00052618, DC MOV Thermal Overload Heater Sizing
AR 00076440, BESS Calculations Self Assessment 50952

Action Requests Written Due to this Inspection

101924, Update periodic maintenance program to add periodic replacement of diaphram in valve E41-PCV-152, dated 08/14/03

102321, Valve E41-F042, reduced voltage strike time calculation basis, dated 08/19/03

102456, CST Vortexing Documentation Discrepancies; dated 08/20/03

103005, Note in 0PT-09.2 Referring to Auto Closure of HPCI Steam Line Drains (F029 and F028) should have been removed by ESR 99-00405, dated 08/26/03

103106, Correct procedure inconsistencies in preventative maintenance Procedure 0PM-BKR001, ITE 4KV Breaker and Compartment Checkout, dated 08/27/03

103252, Procedure Enhancement to 0PT-09.3, Rev. 50, HPCI System - 165 Psig Flow Test. Add Procedural Guidance to Ensure that HPCI Minimum Flow Isolation Valve E41-F012 Goes Closed After Proper Flow Setpoint is Reached, dated 08/28/03

103256, Procedure Enhancement to 0PT-09.2, Rev. 111, HPCI System Operability Test. Add Procedural Guidance to Ensure that HPCI Minimum Flow Isolation Valve E41-F012 Goes Closed After Proper Flow Setpoint is Reached, dated 08/28/03

103299, Provide procedural guidance as to when a Shift Technical Advisor should activate their post, dated 08/28/03

Lesson Plans/Job Performance Measures (JPM)

Lesson Plan CLS-LP-51, DC Distribution, Rev. 0
Lesson Plan CLS-LP-302-G, Electrical Failure Related AOPs (AOP-12.0, AOP-22.0, AOP-36.1, and AOP-39.0), Rev. 0
AOT-OJT-JP-051-A01, DC Ground Isolation for P, N, and P/N, Rev. 1
AOT-OJT-JP-302-G01, Loss of DC Power - Transfer of DC Control Power, Rev. 2

Miscellaneous Documents:

Brunswick Nuclear Plant Probabilistic Safety Assessment
RSC 98-24, Reactor Core Isolation Cooling System Notebook, Rev. 0
RSC 98-23, HPCI System Notebook, Rev. 0
HPCI System Periodic Review, dated 02/20/03
RCIC System Periodic Review, dated 02/20/03
Maintenance Rule Scoping and Performance Criteria, System 1001, ECCS Suction Strainer
Vendor Manual FP-3808, Battery Charger, Rev. G
Specification 137-002, 125 Volt Battery Chargers, Rev. 7
Engineering Evaluation BNP-DC-03, Overload Heater Resizing for Valves 1-E41-F001, F006, F007, and F008, Rev. 0