U. S. NUCLEAR REGULATORY COMMISSION

REGION II

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| Licensee: | Carolina Power & Light Company (CP&L) |
| Facility: | Brunswick Steam Electric Plant, Units 1 & 2 |
| Location: | 8470 River Road<br>Southport, NC 28461 |
| Dates: | January 11 - 15 and January 25 - 29, 1999 |
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# EXECUTIVE SUMMARY

Brunswick Steam Electric Plant
NRC Inspection Report 50-325/98-14, 50-324/98-14

This inspection included a review of the licensee's calculations, analysis, performance test procedures and other engineering activities that were used to support design and performance of the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems during normal and accident or abnormal conditions. The report covered a two-week period of inspection.

Overall, the inspection found that operation of the systems was consistent with the design and licensing basis.

Maintenance

- The maintenance of the HPCI and RCIC systems has been sufficient to support reliable operation of the systems. Maintenance practices have been adequate. Operability of the systems have shown an improved level of performance since mid-1996.

- The material condition of HPCI and RCIC equipment and components examined was good as well as housekeeping in the general areas around equipment and components. This was identified as a strength.

Engineering

- The design control procedures complied with the requirements of 10 CFR 50.59 and 10 CFR 50, Appendix B, Criterion III.

- A violation with two examples was identified for failure to perform 10 CFR 50.59 safety evaluations. A weakness in the licensee's program was identified in the justification for two recently completed 10 CFR 50.59 safety screenings.

- A design control violation example was identified for failure to revise the 1990 calculations that sized the 250 volt DC (VDC) motor operated valve (MOV) thermal overload relay heaters after it was determined that the minimum MCC voltages were significantly lower than had been previously evaluated in the 1990 voltage calculations.

- The design of the HPCI/RCIC electrical components, including control circuits and interfaces with the 125/250 VDC system was consistent with NRC requirements, with the licensing commitments, and with the design bases. The electrical calculation quality was good.

- Instrument setpoint calculations used an approved methodology and considered appropriate sources of instrumentation inaccuracies. Instrumentation surveillance procedures were acceptable and adequate for maintaining the design basis for the HPCI and RCIC systems. Some minor discrepancies were identified in the calculations and procedures.

- A design control violation example was identified for inadequate design of a modification to a minflow valve.

- Design documents and the UFSAR were generally accurate and reflected plant as-built conditions with the exception of the examples identified in two violations. The violations included a failure to update logic drawings in accordance with document control procedures and failure to update the UFSAR in accordance with 10 CFR 50.71(e).

- A design control violation example was identified for failure to translate design requirements into a surveillance procedure for the power uprate project.

- The plant engineering staff was knowledgeable and dedicated to operating the systems as designed. They had a strong sense of ownership and provided good support to operations and maintenance personnel. However, an example was identified wherein the engineering staff did not have a complete understanding of the licensing basis requirements for the HPCI system.

- The licensee has not prepared a DBD for the RCIC system. This may be prudent to do so, since site risk studies show that RCIC is one of three most important risk significant systems.

- The licensee's self-assessment process was effective in identifying problems in program areas. However, long term resolution of deficiencies with 10 CFR 50.59 safety screenings and safety reviews has not yet been demonstrated.

## Report Details

### Introduction

The objective of this Safety System Engineering Inspection (SSEI) was to assess the adequacy of calculations, analysis, other engineering activities, and maintenance practices that were used to support the performance of the HPCI and RCIC systems during normal and accident or abnormal conditions. The inspection was performed by a team of inspectors that included a Team Leader, two Region II Inspectors, and two engineering consultants. Prior to this inspection, the licensee performed an informal review of the design, and licensing basis of the HPCI and RCIC systems. The self assessment results are discussed in Section E7.1, below.

## II. Maintenance

### M2 Maintenance and Material Condition of Facilities and Equipment

#### M2.1 Material Condition of the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems

a. Inspection Scope (IP-93809)

The team reviewed maintenance documentation and conducted walkdown inspections to determine the condition of the high pressure coolant injection (HPCI) and the reactor core isolation cooling (RCIC) systems, and the material condition of the components within the system.

b. Observations and Findings

The team reviewed maintenance documentation and discussed maintenance practices with the HPCI and RCIC system engineer to determine design, maintenance and testing practices, and system performance related to HPIC and RCIC systems. Components included in this review were the suction and discharge piping and valves, steam driven turbines, and main steam supply piping and valves for the turbine driven pumps. The system design, equipment problems encountered, and maintenance practices at Brunswick were also compared to information and industry events described in NRC Information Notices 98-24, 96-68, 96-08, 94-84, 94-27, 94-66, 93-67, 93-51, 88-09 and 86-14 Supplement 1 & 2 to determine if the notices were applicable to Brunswick. The team determined that the licensee had reviewed each of the information notices, and had either completed the appropriate actions or were in the process of completing corrective actions to address each applicable issue. The review also disclosed that due to the design and configuration of components within the HPCI and RCIC systems, many of the reported industry issues were not applicable to Brunswick.

The following maintenance records were reviewed:

- Maintenance work orders from November 1, 1997 thru December 8, 1998 for the HPIC and RCIC systems.

- Maintenance Rule compliance and performance.

- A representative sample of licensee event reports (LERs) from May 1995 thru December, 1998. Corrective actions associated with the LERs were discussed with the system engineer. The team verified the corrective actions had been completed.

In addition the team interviewed licensee engineers and reviewed system operating data to determine whether the HPCI and RCIC systems and the main steam supply lines to the turbine driven pumps had experienced water hammer events, erosion/corrosion problems or service induced discrepancies revealed by inspection. No problems were identified in these areas during this review. The team walked down the accessible portions of the HPCI and RCIC systems to determine the condition of these components. The team noted that material condition of equipment and components examined was excellent as well as housekeeping in the general areas around equipment and components. The system engineer demonstrated a high level of knowledge and familiarity with his assigned systems and was fully aware of industry experience relating to the HPIC and RCIC systems. Based on the reviews performed, the team also noted that since mid-1996 the HPIC and RCIC systems have demonstrated an improved level of performance.

During review of the above records, the following problem was identified: On December 16, 1998, the Unit 2 HPCI system turbine exhaust line vacuum breaker isolation valve, 2-E41-F079, had been placed under clearance to support a scheduled maintenance. Review of the Technical Specifications (TS) and associated bases by operations prior to closing the 2-E41-F079 valve inappropriately determined that closing this valve did not affect HPCI system operability and did not place the HPCI system in a Limiting Condition for Operation (LCO). Subsequent review of the condition by engineering and operations personnel however, determined that the HPCI system had in fact been placed in a condition where it could not meet its design requirements and was declared inoperable. Closure of Valve 2-E41-F079 inhibits the capability of multiple automatic HPIC system starts and stops. This issue was documented and reported to NRC as LER No 2-98-004.

The licensee's Maintenance Rule program states the function of the HPCI system is to provide high pressure ECCS injection to the reactor pressure vessel to maintain water level above the top of the core and prevent ADS actuation for small breaks. The definition of a functional failure for this system injection function states, "inability to deliver 4250 g.p.m. from torus to reactor vessel at pressure from 150 psig to 1164 psig for 8 hours duration." After reviewing LER 2-98-004, the team questioned licensee engineers to determine if this event had been identified as a functional failure and whether this had resulted in the Unit 2 HPCI system to be classified into the Maintenance Rule (a)(1) category. The team determined that the event had not been classified as a functional failure. Licensee engineers immediately realized that guidance given in Regulatory Guide 1.160, Revision 2 clearly stated that, valve mispositioning events associated with maintenance activity should be considered functional failures. In addition, the licensee's maintenance Rule Program Procedure, number ADM-NGGC-0101, defined a functional failure as an unintended condition or event such that a structure, system, or component (SSC) is not capable of performing its intended function. The licensee initiated CR 99-00289, to document and disposition the failure and to consider the event as a functional failure under the maintenance rule (10 CFR 50.65). The functional failure was added to the Unit 2 HPCI injection function. Failure to initially document this event as a functional failure is a violation of 10CFR50.65 which requires performance of appropriate preventive maintenance, such that the SSC remains capable of performing its intended function. This functional failure however did not cause the HPCI to go into Classification (a)(1); was not repetitive; and the licensee's failure to classify was an oversight and not intentional. Licensee engineers took immediate corrective action by initiating a Condition Report (CR) and recording the event as a HPCI functional failure. Therefore, this failure constitutes a violation of minor significance and is not subject to formal enforcement action.

c. Conclusions

The maintenance of the HPCI and RCIC systems has been sufficient to support reliable operation of the systems. Maintenance practices have been adequate. Operability of the systems have shown an improved level of performance since mid-1996. The material condition of HPCI and RCIC equipment and components examined was good, as well as housekeeping in the general areas around equipment and components. This was identified as a strength.

## III. ENGINEERING

### E1 Conduct of Engineering

E1.1 Design Change Control and 50.59 Processes

a. Inspection Scope

The team reviewed the licensee's procedures which control the design change process, including implementation of 10 CFR 50.59 requirements, to determine whether the licensee was adequately controlling the design basis of the plant.

b. Observations and Findings

The team reviewed the current revisions of the licensee's design control procedures. The procedures adequately addressed the following: design input, design verification, control of design output documents, preparation of design calculations, post modification testing, control of field changes, and design engineering training requirements. The procedures provided good controls for maintaining the design basis and for implementation of design changes. Procedures were also reviewed which specified requirements for maintenance of design documents, environmental qualification of electrical equipment, maintenance of the equipment data base system, and review and changes to the UFSAR.

The team reviewed CP&L procedure REG-NGGC-0002, 10 CFR 50.59 and Other Regulatory Evaluations, Revision 1. This procedure implemented interim guidance prepared by NEI to comply with the requirements for performing safety evaluations in accordance with 10 CFR 50.59. The licensee committed to implement the interim guidance for performance of safety evaluations effective July 1, 1998. The procedure provides detailed instructions for performing safety evaluations of temporary and permanent changes to the plant, including procedures. Other regulatory requirements such as fire protection, security, and emergency preparedness were also addressed in procedure REG-NGGC-0002. The procedure requires that all personnel (managers, screeners, and evaluators) involved in preparation and review of safety screens and evaluations be trained and qualified in accordance with the procedure. All safety screenings and safety evaluations are required to be prepared by a qualified individual, be independently reviewed by a qualified reviewer, and be reviewed and approved by a supervisor.

Procedure REG-NGGC-0002 provides detailed instructions for performance of the 10 CFR 50.59 screening which is the initial process for determining if a safety evaluation is required. The initial question in the screening process requires determination if a proposed activity involves a change to the Technical Specifications or operating licensee. If the answer to this question is yes, NRC approval is required before the activity can be implemented. The next series of questions requires determining if the proposed activity involves a change to the facility or procedures as described safety analysis report (SAR), or if the proposed activity involves a test or experiment not described in the SAR. If the answers to any of these questions is yes, a detailed 10 CFR 50.59 safety evaluation is required to determine if the proposed activity could result in an unreviewed safety question (USQ). The procedure requires that answers to questions contain the justification and references in sufficient detail such that another qualified reviewer can independently understand the rationale for the response. The procedure contains explicit instructions regarding considerations for changes to the facility as described in the SAR. A change to a component of any structure, system, or component (SSC) described in the SAR must be evaluated to determine if it affects the design, function, or method of performing the function of a SSC. The impact of a proposed activity for components not described in the SAR on any SSC described in the SAR must also be considered for a USQ determination. This includes changes to components or subcomponents of larger components which may affect the design, function, or method of performing the function of a SSC described in the SAR. The procedure also provides detailed instructions for performance of USQ determinations. Seven specific questions must be answered in the USQ determination. Sufficient detail and references are required for each question answer so that reviewers and subsequent readers are able to reach the same conclusion without having to infer any important information.

c. <u>Conclusions</u>

The design control procedures complied with the requirements of 10 CFR 50.59 and 10 CFR 50, Appendix B, Criterion III.

## E1.2 Electrical Design Review

### a. Inspection Scope

The team examined the 125/250 VDC system and its interfaces with the HPCI/RCIC systems. The team also reviewed electrical control drawings, MOV overload heater sizing calculations, battery load study calculations, battery surveillance testing, and completed electrical modifications on the DC system to determine if the design of the HPCIC/RCIC electrical components, including control circuits, and interfaces with the 125/250 VDC system was consistent with NRC requirements and the licensing and design basis for the systems.

### b. Observations and Findings

The team found that the appropriate HPCI/RCIC loads had been included in the battery load study calculations. The inputs and assumptions used in the calculation for HPCI/RCIC electrical components were reviewed and determined to be reasonable. The battery load study calculations demonstrated that there was adequate capacity in the batteries to supply the design loads for the design duty cycle.

The Technical Specifications require in part that the batteries be load tested to either a service test or performance test profile every refueling outage as appropriate. The performance test, which is required to be performed every five years on the batteries, can be performed in lieu of the service test. The performance test examines battery capacity against the manufacturers rating, while the service test demonstrates the batteries ability to meet the design duty cycle.

The last two load tests performed on the Unit 1 batteries (i.e., one service and one performance test) were reviewed and found to have met test acceptance criteria. The load profile used in the service load test procedures was consistent with that described in the UFSAR. However, the load profile used in the load study calculations for the service load test differed from that shown in the UFSAR. The team discussed the differences with the licensee and they indicated that the licensing and design basis for the batteries was the one minute profile described in the UFSAR. Since the load profiles reflected by the load study calculations were bounded by the UFSAR profiles, the team concluded that use of a different load profile in the calculation did not change the output or conclusions of the calculations. The team found that the acceptance criteria for both the service and performance tests were consistent with the licensing and design basis for the system.

The team reviewed HPCI/RCIC 250 VDC MOV electrical control circuit wiring diagrams. The reviewed control circuit drawings correctly implemented the system operation as described in the licensing and design basis documents for the system. Specifically, the manual and automatic electrical controls for the HPCI 250 VDC motor operated steam admission valve and 250 VDC HPCI pump discharge valve were found to be correct. During review of the control wiring diagrams, the team noted two coils that were labeled on the drawings as "HC". The drawings showed the "HC" coils were wired such that they were in parallel with the motor commutator and armature field when the motor operated in the forward or reverse direction. The team noted that there were no associated contacts shown on the drawing related to these coils. When questioned, licensee engineers were not able to provide any information to the team regarding the function of the "HC" coils in the DC MOV control circuits. The licensee subsequently contacted the vendor of the motor starters, who was unable to provide any additional information on the function of these coils in the motor control circuit. The licensee's followup actions to determine the function of the "HC" coils in the motor control circuits and how they affect the motor overload relay sizing calculations remained open at the end of the inspection. This item will be identified as Inspector Followup Item 50-325,324/98-14-01, Evaluate Function of "HC" Coils in DC MOV Control Circuits.

The team examined the Unit 1 250 VDC safety-related motor operated valves stroke time and motor torque calculation BNP-E-6.109, dated July 31, 1996 and found it to be satisfactory. This calculation determined the minimum and maximum available motor output torques and the valve stroke times at reduced voltage of the 250 VDC safety-related motor operated valves. The results of this calculation were used as inputs in other calculations to determine the acceptability of each valve to perform its safety function and to establish required actuator torque switch settings and limit switch settings.

The team noted that the 1990 calculations that sized the thermal overload relay (TOR) heaters (BNP-E-6.033 and BNP-E-6.032 for Units 1 and 2, respectively) used as inputs "worst case" minimum MCC voltages that were non-conservative as compared to the most recent values identified in the valve stroke time calculation BNP-E-6.109, Revision 1, dated July 31, 1996. The most recently calculated "worst case" minimum MCC voltages were significantly lower than those previously assumed in the 1990 heater sizing calculations. The licensee subsequently failed to evaluate how these lower voltages impacted the thermal overload relay heater sizing calculation results. In response to this issue, the licensee initiated CR BNP 99-00276, on January 27 1999. The licensee immediately performed an assessment of the thermal overload relay heater sizes for the 52 DC MOVs on Units 1 and 2, and concluded that the valves were still operable and capable of performing their safety function. The other licensee corrective actions planned were to revise the appropriate calculations.

The team informed the licensee that this failure to revise the MOV TOR heater sizing calculations (BNP-E-6.033 and -6.032) was contrary to 10 CFR 50, Appendix B, Criterion III, Design Control. Criterion III requires, in part, that changes to design calculations be reviewed to assure they do not affect the design basis or other design documents. This was identified as NCV 50-325,324/98-14-02, Inadequate Control of Design Activities. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Appendix C of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CR 99-00276.

The team reviewed several electrical modification and direct replacement packages that impacted the 125/250 VDC system and found they were completed in accordance with design requirements. The 50.59 Safety Evaluations were considered to be adequate, and no unreviewed safety questions were identified. The specific modifications and direct replacement packages reviewed are listed in an appendix to this report.

c. Conclusions

The design of the HPCIC/RCIC electrical components, including control circuits, and interfaces with the 125/250 VDC system was consistent with NRC requirements, with the licensing basis, and with the design basis for the systems. The electrical calculation quality was good.

A violation example was identified for failure to revise the 1990 calculations that sized the 250 VDC MOV thermal overload relay heaters after it was determined in 1996 that the minimum MCC voltages were significantly lower than had been previously assumed in those calculations.

An IFI was identified to followup on the licensee's review of the function of the "HC" coils in DC motor control circuits.

E1.3 Review of Instrument Setpoint Calculations and Surveillance Procedures

a. Inspection Scope

The team reviewed setpoint calculations and associated surveillance procedures to assure that the plant parameters were being maintained as per the design basis.

b. Observations and Findings

The team reviewed the licensee's setpoint methodology, setpoint calculations and associated surveillance procedures. The team found that setpoint methodology was essentially consistent with the current recommended industry practices. The team noted that many of the calculations were based on General Electric format, often using 'spread sheets'. These spread sheets did not elaborate the derivation of uncertainty/accuracy terms in the calculation. Factors used to derive terms were not documented clearly in the calculations with the spread sheets. However, when a setpoint calculation requires revision, the licensee was converting the calculations into current methodology format which were not dependent on the 'spread sheets'. These revised calculations (for example Calculation 0E41-1002) were more understandable.

The team identified the following minor discrepancies in the calculations and procedures:

- Calculation 0E41-0035, page 12 had discrepancy in scaling figures for 1-E41-PS-N001B. Also Head Correction figures in TECH SPEC ALLOW VALUES in ATTACHMENTS 6, 8 & 9 (for 2E41-PSL-N001A, B & D) of Procedure 0MST-HPCI22Q did not match with those given by Calculation on page 12 and Appendix A. Additionally, the setpoint values between calculation and the procedure did not match. The licensee explained that Revision 2 of the calculation, dated 9/14/98, had been revised but the procedure had not yet been updated. This mismatch would disappear on procedure's oncoming revision.

- Calculation 0E41-0036, page 25 showed relays "E41A-K12.-K32;E51A-K12,-K32" and on page 27 showed HPCI Steam Line Flow - High Time Delay Relay as

"E51-K12,-K32". The correct numbers for HPCI Steam Line Flow - Time Delay Relays should have been E41-K33 & K43.

- Procedure 0PT-09.2 (HPCI System Operability Test) section 7.7.26 item 2 Pump Discharge Pressure referred to gauge as "E51-PI-R001". Gauge should have been "E41-PI-R001".

- The HPCI elementary wiring diagram (for unit 2 - Div. I) 2-FP-50039 sheet 4 showed that steam line high differential pressure switch (Steam Line Break) relay E41-K33 was energized by contact numbers E41-PDTM-N004-1 and E41-PDTS-N004-2. However Calculation number 0E41-0036, Revision 3 for contact number E41-PDTM-N004-1 contained the following statement (on page A-3 of Calculation): "adjusted such that it can never actuate". Therefore, even though the contact number E41-PDTM-N004-1 is shown in the elementary wiring diagram as an active circuit which would close on high differential HPCI steamline pressure, it would never close. The licensee explained that in the original design, a Barton differential pressure switch was used to monitor a high steam flow condition on the HPCI Steam Line. The Barton dp switches were commonly used in an orifice application where it was not uncommon to have flow in either direction. Although the flow occurred in one direction only, the original design adopted the typical orifice configuration and wired both the contacts in the circuit. When the Barton differential pressure switches were replaced with a Rosemount transmitter loop, the negative flow function, which was no longer necessary, was not eliminated. The negative flow condition could never occur because of the orientation of the instrument. Existence of this contact in the circuit had no affect on the operation of ther instrumentation. The licensee does not plan to remove this unnecessary contact from the circuit since system operability was not affected.

The above minor calculation and procedure discrepancies were considered by the team to be examples of failure to pay attention to details. The minor discrepancies did not affect the output of the calculations.

c. Conclusions

Instrument setpoint calculations used an approved methodology and considered appropriate sources of instrumentation inaccuracies. Instrumentation surveillance procedures were acceptable and adequate for maintaining the design basis for the HPCI/RCIC systems. Some minor discrepancies were identified in the calculations and procedures.

## E1.4 Mechanical Design Review

a. Inspection Scope

The team reviewed calculations, design analyses, and surveillance procedures which support the design and licensing basis in the mechanical engineering discipline for the HPCI and RCIC systems. The team also assessed the quality of 10 CFR 50.59 safety evaluations and/or screenings associated with four design modifications, 16 Engineering Service Requests (ESRs), and six Engineering Evaluations to determine whether the licensee was adequately controlling changes to the design basis of the plant .

b. Observations and Findings

HPCI System Response Time Changed From 30 Seconds to 60 Seconds

In June, 1994, the licensee increased the allowable HPCI system response time from 30 seconds to 60 seconds. This change was initiated in UFSAR change number 94FSAR032, dated June 27, 1994. Review of the 10 CFR 50.59 screening which was completed to evaluate the UFSAR change disclosed that the licensee's basis for approval of the change was that the HPCI system was not required for accident conditions. This conclusion was based on NRC's acceptance of the SAFER/GESTAR methodology and analysis results for small break loss of coolant accident (LOCA) with HPCI single failure as permission to remove HPCI from the licensing basis. The NRC issued a Safety Evaluation Report (SER) as an attachment to a letter from NRC to the licensee, dated January 10, 1991, which approved use of SAFER/GESTAR analysis. Based on this SER, the licensee took the position that the HPCI system was not required to be operable.

UFSAR Section 6.3.1.2, "Design Bases", described the HPCI system as, "One high pressure cooling system which is capable of maintaining the water level above the top of the core and preventing automatic depressurization system (ADS) actuation for small breaks". UFSAR Table 6.3.1-1, "Emergency Core Cooling Systems Equipment Design Data Summary", described the HPCI system's design/licensing basis capacity as 4,250 gpm to the reactor vessel over the range of 1,165 psi psid to 150 psid differential pressure between the vessel and primary containment. The operating requirements for the HPCI system are also specified in Technical Specification 3.5.1 and bases.

The 10CFR50.59 safety evaluation for UFSAR change number 94FSAR032 acknowledged that increasing the delay time could allow a small break to actuate the low pressure ECCS systems, thereby bypassing the HPCI function. The basis for acceptability was based on the position that "...in the licensing basis... no credit for the HPCI system is assumed for either small or large breaks." The team concluded that the 10CFR50.59 safety evaluation, which was based on this misinterpretation was inadequate, and the UFSAR change to increase the maximum allowable delay time for HPCI initiation from ≤30 seconds to ≤60 seconds required additional evaluation. The reason for the inadequacy was that the licensee failed to evaluate the effect on water level in the vessel during a small break LOCA which would result from delaying the initiation of HPCI from 30 to 60 seconds. The apparent cause of this error was that licensee engineers did not have a complete understanding of the licensing basis requirements for the HPCI system.

10CFR50.59, "Changes, tests and experiments" required that licensees determine if changes to the facility as described in the SAR could increase the probability of malfunction of equipment important to safety, and thereby involve an unreviewed safety question (USQ). Contrary to this requirement, the licensee increased the allowable time for actuation of the HPCI system for a small break LOCA and did not address the potential that this change could have prevented the system from performing one of its primary licensing basis functions, preventing uncovering of the core. Therefore, the licensee's original safety evaluation was not adequate to meet the requirements of 10CFR50.59. This was identified as a violation of 10 CFR 50.59.

The licensee initiated CR 99-00149 to document and disposition the inadequate safety evaluation. The licensee also initiated CR 99-00157 to document and disposition that the HPCI system response time may have exceeded the licensing basis. Immediate corrective actions were initiated by the licensee to perform testing of components in the Unit 1 and 2 HPCI systems to determine the actual system response. The licensee generated ESR 9900045, Rev 0, dated 1/20/99 to perform testing and analyses of the actual system response times for both units. This work showed that both Units' response times were less than 30 seconds. Therefore, the original licensing basis had not been actually violated.

The licensee initiated ESR 9900062, Rev 0, dated 1/28/99 to re-evaluate the effect of increasing the HPCI system response time from 30 to 60 seconds. This document acknowledged that the original safety evaluation had not addressed the requirement for HPCI to maintain core coverage for small break LOCAs, and it referenced an analysis performed by the vendor, General Electric, which demonstrated that the capacity of the HPCI system could maintain reactor water level for small break LOCAs . ADS would not be actuated and the core would remain covered. Therefore, ESR 99-00062 demonstrated that delaying initiation of the HPCI system response from 30 to 60 seconds did not result in a USQ. The team concluded that the safety evaluation completed as part of ESR 99-00062 complied with the requirements of 10 CFR 50.59.

The violation of 10 CFR 50.59 discussed above is being treated as a Non-Cited Violation, consistent with Appendix C of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CR numbers 99-00149 and 99-00157. This was identified as NCV 50-325, 324/98-14-03, Failure to Perform an Adequate 10 CFR 50.59 Safety Evaluation.

While performing a self-assessment in December, 1998, the licensee identified a discrepancy in UFSAR Table 6.3.3-5, Brunswick ECCS Parameters, and Section 6.3.3.7, Lag Times. The maximum allowed delay time from initiating signal to rated flow available and injection valve wide open were stated to be 30 seconds. These 30 second delay time notations were not revised by the licensee when UFSAR change 94FSAR032 was made. The licensee initiated Condition Report CR 9803013 to document and disposition the UFSAR HPCI response time differences. Review of the CR disclosed that the licensee's proposed corrective actions included revising the UFSAR to change any references for HPCI system response time from 30 seconds to 60 seconds. The justification for the changes was that they were editorial, so that the UFSAR would have a consistent response time (60 seconds) as approved by UFSAR change 94FSAR032

HPCI Check Valve Disk Spring Removal

On 9/24/97, during a routine surveillance inspection, the HPCI turbine exhaust drain pot drain check valve, 2-E41-F022, was found to be missing the piston spring. This valve was a primary containment isolation valve, as defined in UFSAR Table 6.2.4-1 and in the Technical Requirements Manual (TRM), Appendix D, Table 3.6.1.3-1, "Primary Containment Isolation Valves". No spare springs were available at that time, and ESR 97-00575 was generated to allow the piston spring to be an optional component for this valve.

The licensee performed a 10CFR50.59 safety evaluation screening for this design change and judged that a safety evaluation was not required. One of the questions addressed in the screening was, "Does the activity make changes to the facility as described in the SAR?" The licensee responded "no" based on "The FSAR does not address the detailed operation of this valve or its components." The rationale went on to say that, "...the valve is capable of performing its required function satisfactorily with or without the spring." However, neither the screening nor the ESR evaluation addressed the specifics of this valve's "required function", sealing of containment leakage, and the effect that spring removal would have on its ability to seal to the degree required of a containment isolation valve.

The licensee's procedural guidance current at the time the screening was performed, was Procedure 0IA-109, Performance of Nuclear Reviews, Revision 9. Paragraph 3.3.1 of 0IA-109 defined a change to the facility as defined in the SAR as a change to a structure, system, or component (SSC) that is described in the SAR and that may affect the design, function or method of performing an action or process. The individuals responsible for preparation and review of the safety evaluation determined that deletion of the spring did not constitute a change to the facility as described in the SAR. This conclusion was based on fact that the UFSAR did not describe the details of the valve design. However they failed to consider how the change would affect the design or function of the valve.

The failure to perform a 10CFR50.59 safety evaluation for the design change in ESR 97-00575 which allowed removal of a disk spring from containment isolation valve 2-E41-F022 was identified as an additional example of a violation of 10 CFR 50.59, (NCV 50-325, 324/98-14-03). This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Appendix C of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CR 99-00149.

Subsequent to the team's discovery of this discrepancy the licensee performed a 10CFR50.59 safety evaluation for ESR 95-00575. This was documented as Evaluation Identification number 99-0018. The team reviewed the evaluation which provided extensive detailed discussion of the various functions of this valve, its interfaces with the containment and the suppression pool water, how it functioned when HPCI operated, etc. It concluded that removal of the spring would have no affect on the ability of the valve to seal, that this change would not increase the probability of a malfunction of equipment important to safety previously described in the SAR, and therefore, this change did not involve an unreviewed safety question.

RCIC Governor Valve Stem and Seal Modifications

In 1998, two modifications were performed on the RCIC turbine governor valve, E51-V9. The first, under ESR 9800017, Rev 0, dated 3/19/98, changed the valve stem material from nitrided 410 stainless steel to a chromium carbide coated Inconel 718. This change was made to eliminate binding that had been experienced industry-wide with these valves due to a combination of corrosion on the stems and the very close clearances between the stems and the carbon spacer seals. The second change, under ESR98-00477, Rev 0, dated 8/21/98, replaced the carbon spacer seal, with a seal with a larger inside diameter. This was done in response to a 10 CFR Part 21 report from Dresser-Rand, the vendor. Replacement of the spacers was found to be necessary because the new Inconel stem material had a higher thermal expansion coefficient than the original 410 stainless material. At operating temperature this could have resulted in binding as a result of the already close clearances between the valve stem and the carbon spacer seal.

The 10CFR50.59 safety evaluation screenings for both ESRs determined that safety evaluations were not required since the modifications did not require a Technical Specification change, did not change the facility or any procedure as described in the SAR, and did not involve any test or experiment not described in the SAR. Review of the safety evaluation screens disclosed that a part of the documented justification for not requiring a safety evaluation was that the particular components were not described in the UFSAR. For ESR 9800017, the screening stated, in part that "The exact composition of the governor valve stem is not discussed." However, the basis further stated that the new stem will provide the same function as the original stem and is expected to be reliable. The basis also stated that the valve complies with the existing description. For ESR 9800477, the screening stated, "The carbon spacer is a subcomponent of RCIC that is not described in the SAR." The basis further stated that "No change in the RCIC function will occur."

The team determined that although the conclusions reached were correct, i. e. the modifications did not change the facility as described in the SAR, the information on the safety screening documents did not provide sufficient detail to justify the answer to the screening question. This was contrary to the instructions in the applicable licensee procedures (0IA-109, Rev 9 for ESR 9800017 and REG-NGGC-0002 for ESR 9800477, Rev 1) for performance of the safety screenings. This issue was also documented in CR 9900149. The team determined that there was sufficient information in the ESRs to support the conclusions in the screening and therefore these deficiencies were not identified as additional examples of violation item 50-325, 324/98-14-03. However, the failure to provide adequate documentation to support the basis for the 10CFR50.59 safety evaluation screens was identified to the licensee as a weakness in their safety evaluation program. These are similar to other issues with the 10 CFR 50.59 process identified by the licensee during self-assessments, discussed in paragraph E7.1, below.

HPCI Minflow Valve Modification

Modification 89-068, dated 5/18/90, "Replacement of HPCI Globe Valves 1-E41-F008 and 1-E41-F012" replaced both valves to correct problems that had been experienced with the valve operators as well as problems with cavitation due to the high pressure drops that the valves were subjected to during normal operation. Valve number F008 performed the function of throttling flow in the full flow test line back to the condensate storage tank. Valve number F012 was the minimum flow recirculation valve. This valve was required to open to assure that the pump did not operate in a low flow condition that would result in damage to the pump.

Valve F012 was originally a conventional globe-type valve with one large flow path through the valve. It was replaced with a basket-type flow control valve with numerous small openings providing the flow path. Such a design required less operator thrust and it was capable of operating against a high pressure drop without experiencing damaging cavitation.

However, the basket-type design valve for this application was susceptible to plugging from small particles and fibers, such as debris from the suppression pool water. The team determined that since the suppression pool strainer holes were 0.080 inches in diameter and the minimum valve basket flow passages were 0.029 inches in diameter, debris passing through the suppression pool strainer could potentially cause plugging of the basket valve. This could result in failure of the valve to provide the required minimum bypass flow necessary to prevent pump damage. This aspect of the design change was not addressed in any of the modification documents.

The failure to address the potential for plugging of the minimum flow pathway due to the revised design of the new valve was identified as an additional example of a violation of 10 CFR 50, Appendix B, Criterion III, Design Control, (NCV 50-325, 324/98-14-02). A consequence of the inadequate design was that the licensee failed to perform a safety evaluation as required by 10 CFR 50.59. The licensee also failed to translate the design requirements into surveillance procedures to require monitoring of the minimum flow rates during testing to assure detection of the buildup of plugging in the valve which could lead to pump damage. The licensee initiated CR 99-00222 to address the design deficiency. The inadequate 10 CFR 50.59 screening/safety evaluation will be addressed in CR 99-00149.

<u>HPCI/RCIC Drain Pot Drain Line Reroute to Main Condenser</u>

In 1982, the drain lines from the HPCI and RCIC turbine drain pots were modified by rerouting them from the reactor building equipment drain tank to the main condenser. The purpose of the change was to remove a source of high temperature water from the drain tank during normal operation resulting from the discharge of high temperature condensate to the drain tank from the drain pots.

During review of this modification, the team questioned whether this modification circumvented the design intent in that it created what appeared to be new release paths that bypassed secondary containment. Any leakage past the HPCI and RCIC steam line containment isolation valves could potentially proceed unimpeded through this new path. Therefore, this modification appeared to have a potential to increase the consequences of an accident. The safety evaluation that was performed for this modification did not recognize or address this potential.

The licensee responded that this consideration was not required because the Brunswick licensing basis did not require accounting for leakage that bypassed the secondary containment. Discussion of this point led to discovery that bypass leakage was not considered in the licensee's analyses for offsite and control room accident doses.

The potential that these changes could increase the offsite and control room radiation dose consequences for a design basis LOCA had apparently not been considered in design of the modification. The team concluded that additional review of the radiation control aspects of this modification was required. Pending completion of this review, this issue was identified to the licensee as Inspector Follow-up item 50-325, 324/98-14-04, Consideration of Bypass Leakage in Control Room and Offsite Dose Calculations.

Incorrect Technical Specification Bases Descriptions

Technical Specification 3.3.6.1, "Primary Containment Isolation Instrumentation", Table 3.3.6.1-1, Items 3.d and 4.d, described the HPCI and RCIC turbine exhaust diaphragm high pressure isolation instruments respectively. However, the team found that the corresponding Technical Specification Bases described the HPCI and RCIC turbine exhaust (not the exhaust diaphragm) high pressure isolation instruments. Therefore, there was a mismatch between the technical specifications and the bases. The licensee initiated CR 99-00150 to correct these discrepancies in the Bases.

Penetration Flued Heads Design

During review of the environmental qualification aspects of the HPCI and RCIC systems design, the team questioned if the small break LOCA heat loads from the containment penetration flued heads had been considered in the qualification of components in the HPCI/RCIC/RHR penetration room. The function of the flued heads was to prevent the containment structure concrete around hot penetrations, such as main steam, main feedwater, and the HPCI and RCIC steam lines, from exceeding its maximum allowable temperature. Further review of this issue disclosed that the flued heads had been insulated in accordance with specification number 9527-001-249-5, Rev 0, 3/5/82, "Specification for Piping and Equipment Thermal Insulation". Review of drawing number F-01135, Sheet 2, Containment Liner Details, showed that the flued heads were cooled by the reactor building closed cooling water system. Although the presence of insulation resolved the EQ concern, the team questioned whether containment concrete allowable temperatures could be exceeded for certain accident and transient events where reactor building closed cooling water (RBCCW) system cooling water to the penetrations could be lost. In response to this concern, the licensee's subsequent research identified previous communications with the NRC on this subject. PSAR Supplement 2 Comment 5.2.14 and response (no date found) and FSAR Comment 5.10, Amendment 12, and response dated 9/72, addressed the containment concrete temperature concern.

HPCI/RCIC Steam Line Drain Pot Drain Valves Operation

The team identified a concern with the design of the HPCI and RCIC steam line's drain pot drain valves, E41-F028 and F029 for HPCI and E51-F025 and F026 for RCIC. These valves were air-operated and closed on loss of instrument air. The instrument air system was non-safety-related, and therefore not necessarily available under design basis event conditions. Therefore, for those events where the system would be required to cycle on and off, conditions where these valves would be required to open during the idle periods to prevent condensate accumulation in the steam lines, they may not be operable. Such accumulation had the potential to cause turbine overspeed trips and waterhammer, which could prevent the systems from performing their functions. The licensee initiated condition report CR 99-00271 to disposition this issue. This was identified to the licensee as inspector followup item 50-325, 324/98-14-05, HPCI/RCIC Steam Line Drain Valve Operation.

Design of HPCI and RHR Rooms to Prevent Flooding

The design of the plant and equipment arrangement provides for redundancy and physical separation of ECCS systems. Three separate rooms at elevation -17 house the HPCI and RCIC systems, and the low pressure safety injection mode of the RHR system. These are the north RHR, the HPCI, and south RHR rooms. These rooms are designed with water tight doors between them to prevent water from a pipe break in one room from flooding all three rooms simultaneously. The doors are administratively controlled so that they remain closed except when personnel are transiting from one room to another. The team reviewed the design of the floor drain system in the HPCI and RHR rooms to determine if there was a potential that a pipe break in one of the rooms would result in flooding the other rooms as a result of water flowing through the floor drain system. The team reviewed the piping diagrams for the floor drains in these rooms. This review disclosed that each room had a sump and a sump pump and that the floor drains empty into the sump. The floor drain piping systems for each room were not interconnected. The sump pumps discharge to a common header which discharges to the radwaste system. The individual sump discharge lines contain check valves to prevent backflow from one sump pump into another. The check valves are maintained under the maintenance rule.

c. Conclusions

A violation was identified with two examples of failure to perform 10CFR50.59 safety evaluations. Weaknesses were also identified in two recently completed 10 CFR 50.59 safety screenings. A violation was also identified for inadequate design of a modification to the HPCI minflow valve.

The plant engineering staff was knowledgeable and dedicated. They had a strong sense of ownership in the plant and provided good support to operations and maintenance. However, an example was identified wherein the engineering staff did not have a complete understanding of the licensing basis requirements for the HPCI system.

## E3 Engineering Procedures and Documentation

### E3.1 System Design Base Documents

a. Inspection Scope

The team reviewed the Design Basis Document (DBD) for the HPCI system to determine if the DBD was adequate to maintain the design and licensing basis for.

b. Observations and Findings

The current revision of the design basis document (DBD) was Revision 5, dated November 13, 1997. The DBD was a comprehensive document that describes the purpose of the system, system scope and boundaries, interfaces with other systems, functional requirements, design requirements, and the licensing basis for the HPCI system. The design requirements include a listing of controlling calculations, original design codes, and applicable regulatory design criteria. During the pre-inspection self-assessment, the reviewers identified that several DBDs were listed in the reference section of DBD-19 which did not exist. CR 98-093160 was initiated to document and disposition this discrepancy and correct the list of references. One of the non-existent DBDs noted by the NRC was DBD-16 for the RCIC system.

c. Conclusions

DBD-19 was a comprehensive consolidation of the design and licensing basis for the HPCI system. The licensee has not prepared a design basis document for the RCIC system. This may be prudent to do so, since site risk studies show that RCIC is one of the three most important risk significant systems. The lack of a DBD for RCIC could have a negative impact on maintenance of the licensing and design basis for the RCIC system.

### E3.2 Instrumentation & Controls Document Review

a. Inspection Scope

The team reviewed the updated final safety analysis report (UFSAR) and design drawings associated with HPCI and RCIC Systems to assure the correctness of design documents and consistency between the documents.

b. Observations and Findings

The team reviewed the system description, the Design Basis Document (DBD-19), the UFSAR, and design drawings to assure consistency between the documents and to verify that the documents accurately reflected as-built conditions in the plant.

The team noted that the Elementary diagrams had been revised to incorporate various plant modifications, but the logic diagrams (Drawing 0-FP-05482, Sheet 1), had not been updated for the corresponding changes. The component operability was correct because field wiring (as-built conditions) was based on the elementary diagrams which were correct. The team identified the following discrepancy between the elementary wiring diagrams shown on drawing numbers 1-FP-50039, Sheet 7, and 2-FP-50039, Sheet 7 and drawing number 0-FP-05482, General Electric HPCI Functional Control Diagram, Sheet 1. The Unit 2 HPCI steam supply line (motor operated) valve E41-F002 on elementary drawing number 2-FP-50039, Sheet 7, showed that there was no "Seal In" in the opening circuit (i.e. valve could be positioned for throttling). The Unit 1 elementary drawing (1-FP-50039 sheet 7) was similar. However, the logic diagram on drawing number 0-FP-05482, Sheet 1, showed that the signal would "Seal In" in the circuit. The team noted that the elementary diagram was correct as these valves were designed and installed fo: throttling action. The logic drawing (0-FP-05482) was incorrect. Additional discrepancies were also identified in the logic diagrams (drawing number 0-FP-05482) which included discrepancies between the logic diagram for the HPCI FIC POWER LOSS alarm and that shown on HPCI System Elementary Diagram, drawing number 2-FP-50039, Sheet 5; between the logic diagram and that shown for the ERFIS computer inputs on HPCI System Elementary Diagram, drawing number 2-FP-50039, Sheet 3; and between the logic diagram for the override of high steam Line flow signal by switch S35 as shown on HPCI System Elementary Diagram, Drawing number 2-FP-50039, Sheet 4. These same error affected Unit 1 also. The RCIC System logic diagrams had similar errors. CP&L procedure number EGR-NGGC-0007, Revision 3, dated February 26, 1998, requires Category "A" documents to be revised and issued prior to modification turnover. Drawing number 0-FP-05482 was identified in the licensee's document control system as a Category "A" document (drawing). The failure to revise and update Drawing number 0-FP-05482 to incorporate design change information in accordance with procedure EGR-NGGC-0007 was identified to the licensee as a violation of 10 CFR 50, Appendix B, Criterion V, Failure to Revise Drawings to Incorporate Design Change Information (NCV 50-325, 324/98-14-06). This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Appendix C of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CR 99-00166.

During review of the UFSAR, the team identified the following discrepancies:

- UFSAR Figures 7.3.3-2, -3 & -4, HPCI System Functional Control Diagram, had not been updated to reflect changes in the HPCI system logic for modifications to the system where elementary wiring diagrams had been modified. These UFSAR figures did not reflect as built plant configuration and were incorrect, since these figures had not been updated to incorporate the same changes as discussed above for drawing number 0-FP-05482.

- UFSAR Figure 7.3.1-7B showed prefix E51 (in place of E41) for HPCI and the incorrect channel terminal designations of NUMAC cabinet B21-XY-5948A/B for TE-N025C, D, 3488 & 3489.

- Section 7.3.1.1 of the UFSAR did not identify that HPCI turbine steamline would isolate on "HPCI Steam Tunnel Area Temperature" signal, and that RCIC turbine steamline would isolate for "RCIC Steam Line Area Temperature" signal.

Failure to maintain the UFSAR current and accurate was identified to the licensee as a violation of 10 CFR 50.71(e), Failure to Update UFSAR, NCV 50-325, 324/98-14-07. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Appendix C of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CR 99-00219.

c. Conclusions

Design documents and the UFSAR were generally accurate and reflected plant as-built conditions with the exception of the examples identified in two minor violations. The violations included a failure to update logic drawings in accordance with the licensee's document control procedures and a failure to update the UFSAR in accordance with 10 CFR 50.71(e).

E3.3 Consistency of Surveillance Procedures with Design Criteria

a. Inspection Scope

The team reviewed procedures, including engineering process control procedures and surveillance test procedures to verify that the procedures were consistent with the design and licensing basis.

b. Observations and Findings

The team reviewed HPCI pump surveillance test procedures. The following discrepancies were identified:

Step 4.5 of procedure 0PT-10.1.3 required steam supply pressure to be between 135 psig and 165 psig. Technical Specification SR 3.5.3.4 required a turbine inlet pressure between 135 psig and 165 psig. The procedure did not specify which pressure, either reactor pressure or turbine inlet pressure, was required to be in the 135 - 165 psig range during the surveillance test. The licensee initiated CR 99-00192 to clarify the procedure. The team determined that this was not a violation of NRC requirements.

Procedure OPT-09.2, Rev 102, 2/16/98, "HPCI System Operability Test", implemented Technical Specification surveillance requirement SR 3.5.1.7. The procedure's acceptance criteria for pump performance was found to be non-conservative. The procedure required verification that with the reactor pressure ≤ 1,045 psig and ≥ 945 psig the HPCI pump could deliver ≥ 4,250 gpm against a system head corresponding to reactor pressure. At several places in the procedure, this system head was specified as 1,090 psig, including Steps 6.1.1 and 7.7.25, and Attachment 5, Page 2, Item 3. However, Action Item 21 to ESR 95-00238, Rev 0, established that the friction and elevation head losses in the HPCI injection line were 63 psig. Therefore, the minimum system head should have been higher - maximum reactor pressure, 1,045 psig, plus injection line head losses, 63 psig, for a total of 1,108 psig. This would be the pump developed head that would be required to lift the water from the torus (elevation head), overcome system resistance (friction head), and overcome reactor pressure. However, because the test procedure observed pump discharge pressure rather than pump developed head, the team questioned whether the test acceptance criteria should have also been adjusted to account for the static head that would be available during testing from the CST that would not be available when pumping from the torus. The team also questioned whether an allowance for instrument uncertainty was included in the acceptance criteria. The licensee initiated CR 99-00217 to document and disposition this problem. Procedure 0PT-09.2 had not been revised after implementation of the improved Technical Specifications and power uprate, as specified in ESR 95-00238, Revision 0, (Power Uprate) Action Item 21.

Failure to correctly translate the design basis minimum allowable HPCI pump performance into the acceptance criteria for the Technical Specification required operability test procedure, OPT-09.2, Rev 102, 2/16/98, HPCI System Operability Test was identified as an additional example violation of 10CFR50, Appendix B, Criterion III, Design Control, (NCV 50-325, 324/98-14-02). Criterion III, in part, requires that the design basis be correctly translated into procedures and instructions. This violation is in the licensee's corrective action program as CR 99-00217.

c. <u>Conclusions</u>

A violation example was identified for failure to translate design requirements into surveilance procedures for the power uprate project.

**E.7 Quality Assurance in Engineering Activities**

E7.1 Licensee Self Assessments

a. <u>Inspection Scope(37550)</u>

The team reviewed the results of the licensee's informal pre-inspection self-assessment of the HPCI and RCIC systems, and self-assessments performed within the engineering organization.

b. <u>Observations and Findings</u>

The licensee retained three contract engineers to perform an informal self-assessment of the HPCI and RCIC systems prior to this inspection. The results of the self-assessment were documented in an undated report titled HPCI/RCIC Engineering Review. The team reviewed the report. The conclusions from the self-assessment were listed as strengths, findings, and items for management consideration. The findings resulted in initiation of 24 CRs. The findings generally covered the following areas: DBD discrepancies, errors in the UFSAR, calculation discrepancies, ESR discrepancies, and document (drawing or procedure) discrepancies. Most of the document and UFSAR discrepancies were minor. An overall conclusion regarding the calculations was that it was sometimes difficult to determine the calculation of record. None of the findings resulted in any operability issues.

The team also reviewed the results of six self-assessments performed within the engineering organization during 1998. Subjects covered by the self-assessments included the following areas: local leak rate program, cooling water systems erosion/corrosion monitoring program, corrective action program, Control Building HVAC, safety relief valve certification program, and the 10 CFR 50.59 program. The self-assessments were effective in identification of issues. The 10 CFR 50.59 program self-assessment was performed from October 5 through 16, 1998, to evaluate the quality of safety reviews performed in accordance with REG-NGGC-0002 following implementation of this procedure on June 30, 1998. Similar findings were identified by this engineering self-assessment as identified in the site wide 10 CFR 50.59 self-assessment discussed below.

In addition, the team reviewed a self-assessment of the site wide implementation of the 10 CFR 50.59 program which was performed by the site Regulatory Compliance organization from August 22 to September 17, 1998. The purpose of the self-assessment was to evaluate the quality of safety reviews performed in accordance with the new corporate procedure, REG-NGGC-0002, which was implemented on June 30, 1998. Safety reviews associated with 35 procedure changes and 7 ESRs under the new procedure (REG-NGGC-0002) were reviewed during the self-assessment. The findings of the self-assessment were that safety reviewers did not fully understand the requirements for performance of safety reviews in that 24 of 42 of the safety reviews contained varying degrees of deficiencies. Most of the deficiencies were considered administrative in nature, although one technical deficiency was identified which resulted in initiation of CR 98-02333. This safety evaluation involved changes to the turbine building closed cooling water (TBCCW) outlet heat exchanger temperature which failed to revise the TBCCW temperature. None of the deficiencies resulted in an incorrect conclusion regarding the USQ determination, or an inadequate safety evaluation. The conclusions of the self-assessment were that additional training was required for reviewers to improve the quality of screenings and safety evaluations, primarily with the emphasis on documentation of references and justifications of conclusions. In addition, the self-assessment identified that procedure REG-NGGC-0002 required revision to clarify and simplify the 10 CFR 50.59 process.

The team also reviewed NAS Assessment Report No. B-SP-97-06, Brunswick 50.59 Safety Review Program Assessment. This assessment was performed from December 8 - 17, 1997. Two issues and three items for management consideration were identified by the NAS assessment. One of the issues concerned lack of management involvement in the 50.59 process in that they failed to provide quality standards or implement adequate performance monitoring to ensure program guidance. The other issue identified that safety reviews were of poor quality and did not meet high standards. The items for management consideration concerned administrative issues which were by the new corporate procedure (REG-NGGC-0002) and implementation of the procedure onsite.

The findings from the assessments and this inspection indicated that in the 10CFR50.59 screenings, justification and documentation for the answer to the question, "Was the SSC described in the SAR?", often focused on whether the specific item being modified was described in the SAR. The answer should have documented whether or not the change affected the function of the SSC. The licensee's corrective actions will include additional training in improving the quality of safety screenings and safety evaluations.

c. Conclusions

The licensee's pre-inspection self-assessment was effective in identifying several issues which were addressed in the corrective action program. The self-assessment program was effective in identifying problems in program areas. However resolution of deficiencies with 10 CFR 50.59 safety screenings and safety reviews has not yet been effective.

## V. MANAGEMENT MEETINGS

### X1 Exit Meeting Summary

The Team Leader discussed the progress of the inspection with licensee representatives on a daily basis and presented the results to members of licensee management and staff at the conclusion of the inspection on January 29, 1999. The licensee acknowledged the findings presented.

### PARTIAL LIST OF PERSONS CONTACTED

Licensee

W. Dorman, Manager, Licensing and Regulatory Affairs
J. Franke, Superintendent, Mechanical Engineering, Brunswick Engineering Support Section (BESS)
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M. Grantham, Supervisor, Mechanical/Civil Design, BESS
E. Hux, Director, Site Operations
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J. Titrington, Supervisor, ECCS Systems, BESS
S. Vann, Superintendent, Technical Services, BESS
H. Willets, Electrical/I&C Systems, BESS,
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Other licensee employees contacted included engineers, Nuclear Assessment personnel and administrative personnel.

NRC

B. Mallet, Director, Division of Reactor Safety
T. Easlick, Senior Resident Inspector
E. Brown, Resident Inspector
G. Guthrie, Resident Inspector

## LIST OF INSPECTION PROCEDURES USED

IP 93809 Safety System Engineering Inspection

## LIST OF ITEMS OPENED

| Item Number | Type | Status | Description and Reference |
|---|---|---|---|
| 50-325,324/98-14-01 | IFI | Open | Evaluate Function Of HC coils in DC MOV Control Circuits heater sizing calculations (Section |
| 50-325, 324/98-14-02 | NCV | Closed | Inadequate Control of Design Activities |
| 50-325, 324/98-14-03 | NCV | Closed | Failure to Perform an Adequate10 CFR 50.59 Safety Evaluations |
| 50-325(324)/98-14-04 | IFI | Open | Consideration of Bypass Leakage in Control Room and Offsite Dose Calculations |
| 50-325(324)/98-14-05 | IFI | Open | HPCI/RCIC Steam Line Drain Valve Operation |
| 50-325(324)/98-14-06 | NCV | Closed | Failure to Revise Drawings to Incorporate Design Changes |
| 50-325(324)/98-14-07 | NCV | Closed | Failure to Update UFSAR |

APPENDIX 1

## LIST OF DOCUMENTS REVIEWED

### TECHNICAL SPECIFICATIONS

3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation

3.3.5.2 Reactor Core Isolation Cooling (RCIC) System Instrumentation

3.3.6.1 Primary Containment Isolation Instrumentation

3.5.1 ECCS - Operating

3.5.2 ECCS - Shutdown

3.5.3 RCIC System Unit 1 Technical Specification 3.5 and Bases, Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System

Unit 1 Technical Specification 3.3 and Bases, Instrumentation

Unit 1 Technical Specification 5.5.2, Primary Coolant Sources Outside Containment

### UFSAR

UFSAR Section 6.3, Emergency Core Cooling System (ECCS)

UFSAR Section 5.4.6, Reactor Core Isolation Cooling System (RCIC)

UFSAR Section 7.3.1.1.6.7, Reactor Core Isolation Cooling (RCIC) Equipment Area and Steam Line Tunnel High Temperature and High Differential Temperature

UFSAR Section 7.3.1.1.6.8, RCIC Turbine High Steam Flow

UFSAR Section 7.3.1.1.6.9, RCIC Turbine Steam Line Low Pressure

UFSAR Section 7.3.1.1.6.10, RCIC Turbine Exhaust High Pressure

UFSAR Section 7.3.1.1.6.11, HPCI Equipment Area and Steam Line Tunnel Area High Temperature and HPCI Steam Line Tunnel Area High Differential Temperature

UFSAR Section 7.3.1.1.6.12, HPCI Turbine High Steam Flow

UFSAR Section 7.3.1.1.6.13, HPCI Turbine Steam Line Low Pressure

UFSAR Section 7.3.1.1.6.14, HPCI Turbine Exhaust High Pressure

UFSAR Section 7.3.1.1.9.3, Reactor Core Isolation Cooling System and High Pressure Coolant Injection System

UFSAR Section 7.3.3, Core Standby Cooling Systems

UFSAR Section 7.4, Systems Required for Safe Shutdown

UFSAR Section 15.2.1, Generator Load Rejection

UFSAR Section 15.2.2, Turbine Trip

UFSAR Section 15.2.3, Main Steam Isolation Valve Closure

UFSAR Section 15.2.4, Loss of Condenser Vacuum

UFSAR Section 15.2.5, Loss of Auxiliary Power

UFSAR Section 15.2.6, Loss of Feedwater Flow

## CALCULATIONS

BNP-E-6.033, Rev. 1, dated June 19, 1990, DC Valve Overload Relay Heater Sizing (Unit 1)

BNP-E-6.074, Rev. 0, dated December 16, 1994, Unit 1 - 125/250V DC Battery Load Study

BNP-E-6.109, Rev. 1, dated July 29, 1996, Stroke Time and Motor Torque Calculation for 250 VDC Safety-Related Motor-Operated Valves

BNP-MECH-E41-F001, Rev. 1, dated November 6, 1997, Mechanical Analysis and Calculations for 1/2-E41-F001 HPCI Turbine Steam Admission Valve

BNP-MECH-E41-F004, Rev. 0, dated November 12, 1997, Mechanical Analysis and Calculations for 1/2-E41-F004 HPCI Condensate Storage Tank Suction Valves

BNP-MECH-E41-F008, Rev. 0, dated October 10, 1997, Mechanical Analysis and Calculations for 1/2-E41-F008 HPCI Bypass to CST Valves

BNP-MECH-E41-F006, Rev. 2, dated April 24, 1998, Mechanical Analysis and Calculations for 1/2-E41-F006 High Pressure Coolant Injection Valve

BNP-MECH-E51-F008, Rev. 1, dated February 27, 1998, Mechanical Analysis and Calculations of 1/2-E51-F008 RCIC Steam Supply Outboard Isolation Valves

BNP-MECH-E51-F029, Rev. 0, dated October 24, 1997, Mechanical Analysis and Calculations for 1/2-E51-F029 RCIC Outboard Suppression Pool Suction Valves

BNP-MECH-E51-F031, Rev. 0, dated November 3, 1997, Mechanical Analysis and Calculations for ½-E51-F031 RCIC Inboard Suppression Pool Suction Valves

BNP-MECH-E51-F022, Rev. 0, dated January 19, 1998, Mechanical Analysis and Calculations for 1/2-E51-F022 RCIC Test Bypass Valve

BNP-MECH-E51-F045, Rev. 0, dated October 10, 1997, Mechanical Analysis and Calculations for 1/2-E51-F045 RCIC Turbine Steam Admission Valves

BNP-MECH-E51-F007, Rev. 2, dated March 24, 1998, Mechanical Analysis and Calculations for 1/2-E51-F007 RCIC Steam Supply Inboard Isolation Valves

0E41-1002 Rev. 1, High Pressure Coolant Injection System - Suppression Pool Water Level - High Uncertainty and Setpoint

Calculation (For HPCI, 1(2)E41-LSH-N015A(B))

0E41-0036 Rev. 3, Power Uprated HPCI Steamline Flow High Uncertainty and Scaling Calculation (HPCI E41-N004,-005)

0E41-0035 Rev. 2, HPCI Steam Supply Pressure Low Uncertainty and Scaling Calc (E41-PS-N001A-D)

0E41-0037 Rev. 3, Power Uprated HPCI Turbine Exhaust Diaphragm Pressure High Uncertainty and Scaling Calculation

0E41-1001 Rev. 0, High Pressure coolant Injection System - Condensate Storage Tank Level - Low Uncertainty and Setpoint Calculation (For HPCI, 1(2) E41-LSL-N002(3))

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## PROCEDURES

ADM-NGGC-0101, Rev. 10, Maintenance Rule Program

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EGR-NGGC-0003, Design Review Requirements

EGR-NGGC-0005, Rev. 9, Engineering Service Request

EGR-NGGC-0007, Rev. 3, Maintenance of Design Documents

EGR-NGGC-0106, Rev. 1, AC and DC Overcurrent Protection and Coordination

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0AP-024, Rev. 1, Development, Review, and Approval of Licensing Document Changes(UFSAR)

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0MST-BATT11R, Rev. 3, dated June 17, 1998, Batteries, 125 VDC, Service Capacity Test

0MST-BATT12R, Rev. 3, dated July 10, 1998, Batteries, 125 VDC, Operability Test

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0MST-HPCI26Q Rev. 2, HPCI Suppression Pool High Level Instrument Chan Cal

2MST-HPCI21Q Rev. 4, HPCI Steam Line Break High D/P Trip Unit Chan Cal

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0MST-HPCI22Q Rev. 1, HPCI Steam Line Low Press Inst Chan Cal

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OPT-09.7, Rev. 2, dated November 16, 1998, HPCI System Valve Operability Test

OPT-10.1.1, Rev. 77, dated February 16, 1998, RCIC System Operability Test

OPT-10.1.3, Rev. 44, dated October 2, 1997, RCIC System Operability Test - Flow Rates at 150 psig

OPT-20.2, Rev. 19, dated August 14, 1998, Section XI Leak Testing

## MODIFICATIONS

PM 88-015, DC Motor Surge Suppression

PM 82-030, 125V Battery Charger Overvoltage Protection

PM 84-005, Install Alternate 480 VAC Feed to Battery Charger 1B-1 From MCC 1XB Compt D3A and Associated Transfer Switch

PM 84-007, Providing Alternate 480 VAC Source and Associated Transfer Switch for 125/250 VDC Battery Charger 1B-2

PM 86-011, Replace Existing 37.5 KVA UPS System with New 50 KVA Equipment

PM 92-079, HPCI & RCIC TOPAZ Inverter Replacement

PM 92-080, HPCI & RCIC TOPAZ Inverter Replacement

PM 92-131, DC Ground Detection

DR 90-0106, Direct Replacement of 125 VDC Class 1E Plant Battery 2A-1

89-030 HPCI/RCIC Reliability Improvement

85-087 E41A-TDR-K33 & K43 Setpoint Change

89-068, Rev. 0, dated October 19, 1990, Replacement of HPCI Globe Valves 1-E41-F008 and 1-E41-F012

ESR 9500938, Rev. 0, dated February 15, 1996, HPCI Exhaust Drain Pot Valve, 2-E41-F053, Disk Notch Modification

ESR 9500711, Rev. 0, dated August 18, 1995, Remove Valve 2-E51-F002

82-138, Rev. 0, dated February 15, 1984, HPCI/RCIC Steam Trap & Valve Upgrade

## ENGINEERING SERVICE REQUESTS (ESRs)

95-0238, Rev. 0, dated May 16, 1996, Perform a calculation for the line losses in the HPCI system

96-0393, Rev. 0, dated May 25, 1996, Provide operability determination for HPCI and RCIC Testing

9501063, Rev. 0, dated June 27, 1995, Evaluate Undersized HPCI Studs

9600373, Rev. 0, dated June 12, 1996, SIL No. 597, HPCI System Injection Capability

9501046, Rev. 0, dated August 25, 1995, Seal Detail Different Than Seal Installed in Field

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98-00017, Rev. 0, dated March 19, 1998, RCIC Governor Valve Stem Material Evaluation

9900045, Rev. 0, dated January 20, 1999, HPCI Response Time Determination

970575, Rev. 0, dated September 30, 1997, Acceptability of Using 2-E41-F022 Without Piston Spring

9600359, Rev. 1, dated July 26, 1996, Evaluate Classification of CIVS

96-00639, Rev. 0, dated November 2, 1996, Evaluation of HPCI Drain Pot Valves E41-F028 & E41-F029

9900045, Rev. 0, dated January 20, 1999, HPCI Response Time Determination

9900062, Rev. 0, dated January 28, 1999, HPCI Response Time Re-Evaluation

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## ENGINEERING EVALUATIONS

94-0112, Rev. 0, dated June 17, 1994, Unit 2 HPCI & RCIC 300% Steam Flow Isolation Instrumentation Evaluation

94-0058, Rev. 0, dated March 9, 1994, Use of Auxiliary Steam for Unit 2 RCIC/HPCI Low Pressure Testing

94-0007, Rev. 0, dated March 7, 1994, HPCI and RCIC Keepfill Settings Change Evaluation

92-0280, Rev. 0, dated September 12, 1992, Keepfill Station Improvements

91-0116, Rev. 0, dated September 24, 1991, Increased RCIC Injection Flow

88-0295, Rev. 0, dated July 29, 1988, Evaluation of Potential Safety-Related Pump Loss

SP-88-026, 88-025, Estimate Choked Flow After a HPCI or RCIC Steam Line Break

## DRAWINGS

D-02543, Sheets 1A, Rev 41, & 1B, Rev 37, Reactor Building Piping Diagram

D-02544, Rev 24, Reactor Building Piping Diagram

D-25023, Sheet 1, Rev. 53, dated December 3, 1998, Piping Diagram, High Pressure Coolant Injection System, Unit 1

D-25023, Sheet 2, Rev. 42, dated December 3, 1998, Piping Diagram, High Pressure Coolant Injection System, Unit 1

D-25029, Sheet 1, Rev. 51, dated December 3, 1998, Piping Diagram, Reactor Core Isolation Cooling System, Unit 1

D-25029, Sheet 2, Rev. 38, dated December 3, 1998, Piping Diagram, Reactor Core Isolation Cooling System, Unit 1

2-FP-82821, Rev. A, dated November 4, 1992, Valtek Certified Dimensional Drawing

F-01135, Rev. 27, dated August 11, 1994, Containment Liner Details - Sheet No. 2

D-02543, Sh 1A, Rev. 41, dated February 28, 1996, Reactor Building Piping Diagram Bldg. Northeast CRW Drainage

9527-F-4075, Rev. 13, dated April 5, 1994, Radwaste Building Ventilation System

0-FP-05482, Sheet 1 of 3, Rev. D, High Pressure Coolant Injection Unit 1 & 2 [FCD]

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0-FP-05482, Sheet 3 of 3, Rev. A, High Pressure Coolant Injection Unit 1 & 2 [FCD]

2-FP-50039, Sheet 1, Rev. L, HPCI System Elementary Diagram Unit 2

2-FP-50039, Sheet 2, Rev. M, HPCI System Elementary Diagram Unit 2

2-FP-50039, Sheet 3, Rev. U, HPCI System Elementary Diagram Unit 2

2-FP-50039, Sheet 4, Rev. P, HPCI System Elementary Diagram Unit 2

2-FP-50039, Sheet 5, Rev. H, HPCI System Elementary Diagram Unit 2

2-FP-50039, Sheet 6, Rev. P, HPCI System Elementary Diagram Unit 2

2-FP-50039, Sheet 7, Rev. E, HPCI System Elementary Diagram Unit 2

1-FP-50039, Sheet 1, Rev. H, HPCI System Elementary Diagram Unit 1

1-FP-50039, Sheet 2, Rev. J, HPCI System Elementary Diagram Unit 1

1-FP-50039, Sheet 3, Rev. U, HPCI System Elementary Diagram Unit 1

1-FP-50039, Sheet 4, Rev. Q, HPCI System Elementary Diagram Unit 1

1-FP-50039, Sheet 5, Rev. L, HPCI System Elementary Diagram Unit 1

1-FP-50039, Sheet 6, Rev. Q, HPCI System Elementary Diagram Unit 1

1-FP-50039, Sheet 7, Rev. E, HPCI System Elementary Diagram Unit 1

1-FP-50039, Sheet 8, Rev. H, HPCI System Elementary Diagram Unit 1

0-FP-05548, Sheet 1 of 3, Rev. C, RCIC System Functional Control Diagram Unit 1 & 2

FP-9527-5548, Sheet 2 of 3, Rev. 3, RCIC System Functional Control Diagram Unit 1 & 2

FP-9527-5548, Sheet 3 of 3, Rev. 4, RCIC System Functional Control Diagram Unit 1 & 2

2-FP-50098, Sheet 1, Rev. M, RCIC System Elementary Diagram Unit 2

2-FP-50098, Sheet 2, Rev. R, RCIC System Elementary Diagram Unit 2

2-FP-50098, Sheet 3, Rev. U, RCIC System Elementary Diagram Unit 2

2-FP-50098, Sheet 4, Rev. M, RCIC System Elementary Diagram Unit 2

2-FP-50098, Sheet 5, Rev. S, RCIC System Elementary Diagram Unit 2

2-FP-50098, Sheet 6, Rev. G, RCIC System Elementary Diagram Unit 2

2-FP-50098, Sheet 7, Rev. C, RCIC System Elementary Diagram Unit 2

1-FP-50098, Sheet 1, Rev. Q, RCIC System Elementary Diagram Unit 1

1-FP-50098, Sheet 2, Rev. Q, RCIC System Elementary Diagram Unit 1

1-FP-50098, Sheet 3, Rev. U, RCIC System Elementary Diagram Unit 1

1-FP-50098, Sheet 4, Rev. H, RCIC System Elementary Diagram Unit 1

1-FP-50098, Sheet 5, Rev. T, RCIC System Elementary Diagram Unit 1

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1-FP-50098, Sheet 7, Rev. D, RCIC System Elementary Diagram Unit 1

F-30008, Rev.29, dated October 28, 1998, Unit 1 Three Line Diagram 125/250 Volt DC System MCC-1XDA, 1XDB,1TDA,1TDB

F-30049, Rev.89, dated November 19, 1998, Unit 1 Auxiliary One Line Diagram 480V System MCC-1XA, 1XC, 1XE, 1XG, 1XJ, 1XL & 1XA-2

LL-92072, SH.23, Rev. 13, dated May 11, 1998, Unit 1 MCC "1XDA" COMPT "B17" HPCI Pump Discharge Valve 1-E41-F006 Control Wiring Diagram

LL-92072, SH.31, Rev.11, dated November 2, 1996, Unit 1 MCC "1XDA" COMPT "B21" HPCI Steam Supply Valve to Turbine 1-E41-F001 Control Wiring Diagram

## INFORMATION NOTICES

Notice 98-24, Dated June 26, 1998, Steam Binding in Turbine Governor Valves in Reactor Core Isolation Cooling (RCIC) and Auxiliary Feedwater (AFW) Systems

Notice 96-68, Dated December 19, 1996, Incorrect Effective Diaphragm Area Values in Vendor Manual Resulted in Potential Failure of Pneumatic Diaphragm Actuators

Notice 96-08, Dated February 5, 1996, Thermally Induced Pressure Locking of a High Pressure Coolant Injection Gate Valve

Notice 94-84, Dated December 2, 1994, Air Entrainment in Terry Turbine Lubricating Oil System

Notice 94-66, Dated June 16, 1995, Overspeed of Turbine-Driven Pumps Caused by Binding in Stems of Governor Valves, Supplement 1

Notice 94-66, Overspeed of the Turbine-Driven Pumps Caused by Governor Valve Stem Binding

Notice 94-27, Dated March 31, 1993, Facility Operating Concerns Resulting from Local Area Flooding

Notice 93-67, Dated August 16, 1993, Bursting of High Pressure Coolant Injection Steam Line Rupture Disc Injures Plant Personnel

Notice 93-51, Dated July 9, 1993, Repetitive Overspeed Tripping of Turbine-Driven Auxiliary Feedwater Pumps

Notice 88-09, Dated April 18, 1988, Reduced Reliability of Steam-Driven Auxiliary Feedwater Pumps Caused by Instability of Woodard PG-PL Type Valves

Notice 86-14, Dated August 26, 1986, Overspeed Trips of AFW, HPCI, and RCIC Turbines, Supplement No.2

Notice 86-14, Dated December 17, 1986, Overspeed Trips of AFW, HPIC, and RCIC Turbines, Supplement No 1

## LICENSEE EVENT REPORTS

LER No. 2-98-004, High Pressure Coolant Injection Rendered Inoperable

LER No. 2-98-001, Reactor Core Isolation Cooling System Isolation Instrumentation Setpoint Shift

LER No. 1-97-013, Reactor Core Isolation Cooling System Surveillance Procedure Inadequacy

LER No. 2-97-003, High Pressure Coolant Injection Inoperability - Installation of Non-Seismically Supported Temporary Air Piping

LER No. 1-96-003, HPCI Valve Body Provisions for Bonnet and Yoke did not Insure Valve Internals were Concentric

LER No. 1-96-006, RCIC Surveillance Procedure Inadequate

LER No. 2-95-002, Failed Resistor in HPCI System Power Circuit Supply

LER No. 1-95-022, Abnormal HPCI Turbine Operation Due to Inadequate Flushing of Hydraulic Operator Following Maintenance Activities

LER No. 1-95-013, Ground Associated with the HPCI Barometric Condenser Vacuum Pump Resulted in a Ground on the "A" Battery Bus

LER No. 2-97-003, High Pressure Coolant Injection Inoperability - Installation of Non-Seismically Supported Temporary Air Supply.

LER No. 1-97-013, Reactor Core Isolation Cooling System Surveillance Procedure Inadequacy

LER No. 2-98-001, Reactor Core Isolation Cooling System Isolation Instrumentation Setpoint Shift

LER No. [illegible]-96-006, Technical Specification Surveillance Acceptance Criteria Did Not Adequately Account for Head Losses

**Condition Reports**

CAPS 96-01675, Inadequate HPCI/RCIC Operability Test Acceptance Criteria

CAPS 98-01780, 7/16/98, UFSAR Discrepancy Regarding ADS Operation

CAPS 98-00343, 2/12/98, CST Volume Description Errors

CAPS 98-03116, 9/16/97, Actuator Torque Too High

CAPS 97-01758, 5/15/97, RCIC System Leakage/Spill

CAPS 97-00738, 2/18/97, UFSAR 6.2.3 Discrepancy

BNP 99-00217, 1/21/99, Lack of PT-09.2 ITS Update

CAPS 96-01675, 5/24/96, POT-10.1.1 Acceptance Criteria

BNP 99-00277, 1/27/99, Flued Head Insulation

BNP 99-00271, 1/27/99, Should E41-F028/29 Fail Closed

**MISCELLANEOUS DOCUMENTS**

DBD-19, Rev.5, dated November 13, 1997, HPCI System

DBD-50, Rev.2, dated November 6, 1997, AC Electrical System

DBD-51, Rev.4, dated October 27, 1998, DC Electrical System

Specification No. 249-002, Rev 13, 4/6/98, Specification for Thermal Insulation of Piping and Equipment

Specification No. 9527-01-249-2, 3/25/75, Specification for Thermal Insulation of Piping and Equipment

Battery Surveillance Test Work Request/Job Order Records

WR/JO ALKW002, WR/JO ALKW003, WR/JO ALKX002, WR/JO ALKX003, WR/JO ALKY002, WR/JO ALKY003, WR/JO ALKZ002, WR/JO ALKZ003, WR/JO ANTK001, WR/JO ANST001, and WR/JO ANSN001