April 25, 2001

Mr. Ted C. Feigenbaum
Executive Vice President and Chief Nuclear Officer
Seabrook Station
North Atlantic Energy Service Corporation
c/o Mr. James M. Peschel
P.O. Box 300
Seabrook, NH 03874

SUBJECT: SEABROOK NRC INSPECTION REPORT 05000443/2001-004

Dear Mr. Feigenbaum:

On March 23, 2001, the NRC completed a team inspection of the residual heat removal system and the evaluation of changes, tests, and experiments at your Seabrook facility. The enclosed report presents the results of that inspection. The preliminary findings were discussed with Messrs. G. St. Pierre and J. Vargas and members of your staff on March 23, 2001.

The inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with conditions of your license. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

The team identified one issue of very low safety significance (Green) that involved a violation of NRC requirements. The finding relates to design input errors associated with engineering calculations. However, because of its very low safety significance and because the issue has been entered into your corrective action program, the NRC is treating this issue as a non-cited violation in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you contest this non-cited violation, 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 I, the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001, and the NRC Resident Inspector at the Seabrook Station.

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

Sincerely,

/RA/

Wayne D. Lanning, Director
Division of Reactor Safety

Docket No. 05000443
License No. NPF-86

Enclosure: NRC Inspection Report No. 05000443/2001-004

Attachments:
(1) Supplemental Information
(2) List of Documents Reviewed
(3) List of Acronyms

cc w/encl:
B. D. Kenyon, President and Chief Executive Officer
J. M. Peschel, Manager - Regulatory Programs
G. F. St. Pierre, Station Director - Seabrook Station
D. G. Roy, Nuclear Training Manager - Seabrook Station
D. E. Carriere, Director, Production Services
W. J. Quinlan, Esquire, Assistant General Counsel
W. Fogg, Director, New Hampshire Office of Emergency Management
D. McElhinney, RAC Chairman, FEMA RI, Boston, Mass
R. Backus, Esquire, Backus, Meyer and Solomon, New Hampshire
D. Brown-Couture, Director, Nuclear Safety, Massachusetts Emergency Management Agency
F. W. Getman, Jr., Vice President and Chief Executive Office, BayCorp Holdings, LTD
R. Hallisey, Director, Dept. of Public Health, Commonwealth of Massachusetts
M. Metcalf, Seacoast Anti-Pollution League
D. Tefft, Administrator, Bureau of Radiological Health, State of New Hampshire
S. Comley, Executive Director, We the People of the United States
W. Meinert, Nuclear Engineer
S. Allen, Polestar Applied Technology, Incorporated
R. Shadis, New England Coalition Staff

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REGION I

Docket/Report No:	05000443/2001-004
License No:	NPF-86
Licensee:	North Atlantic Energy Service Corporation (NAESCO)
Facility:	Seabrook Generating Station, Unit 1
Location:	P.O. Box 300 Seabrook, New Hampshire 03874
Dates:	March 7-9 and 19-23, 2001
Inspectors:	L. Prividy, Senior Reactor Inspector, Team Leader, DRS R. Bhatia, Reactor Inspector, DRS D. Dempsey, Reactor Inspector, DRS M. Ferdas, Reactor Inspector, DRS A. Lohmeier, Senior Reactor Inspector, DRS E. Harold Gray, Senior Reactor Inspector, DRS
Approved by:	Lawrence T. Doerflein, Chief Systems Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 050000443-01-04; on 3/7-9, 19-23/2001; North Atlantic Energy Service Corporation; Seabrook Generating Station; Mitigating Systems; Evaluations of Changes, Tests, or Experiments; Other Activities (PI&R).

The inspection was conducted by a region-based team of the residual heat removal (RHR) system using NRC Baseline Inspection Procedure 71111.21, "Safety System Design and Performance Capability." The team also reviewed the conduct of evaluations of changes, tests and experiments under the 10 CFR 50.59 process using NRC Baseline Inspection Procedure 71111.02, "Evaluations of Changes, Tests, and Experiments." The significance of issues is indicated by their color (Green, White, Yellow, or Red) and was determined by the Significance Determination Process (SDP) in Inspection Manual Chapter 0609.

Cornerstone: Mitigating Systems

- Green. The team identified two design input errors in engineering calculations which led to non-conservative results associated with a plant design change and an instrument setpoint. A design input error for refueling water storage tank (RWST) level used in a pump performance calculation resulted in the use of non-conservative emergency core cooling system pump flow rates to support design change DCR 00-013, "Manual Transfer to Cold Leg Recirculation." Also, the failure to include a loop uncertainty allowance for the RWST contained volume water level instrument resulted in a non-conservative alarm setpoint. These design input errors were a violation of 10 CFR 50, Appendix B, Criterion III, Design Control. The issue was a non-cited violation and was entered into the licensee's corrective action program.

  The conditions associated with this violation were determined to be of very low safety significance since compensating margins existed and, when applied to the calculation results, provided assurance of system functionality. (Section 1R21.1)

Report Details

1. REACTOR SAFETY

Cornerstones: Mitigating Systems and Barrier Integrity

1R21 Safety System Design and Performance Capability (IP 7111121)

Introduction

The team selected the residual heat removal (RHR) system for its review of the design and performance capability of safety systems at the Seabrook plant. The system was selected because of its risk significance in event mitigation, barrier integrity, and core damage prevention. Under normal operating conditions, the RHR system transfers heat from the reactor coolant system to the primary component cooling system during the second phase of plant cooldown. Under accident conditions, the primary function of the RHR system is to prevent excessive fuel cladding temperatures by providing borated low pressure coolant to the reactor core in the event of a large break loss-of-coolant accident (LOCA). The team particularly focused on system operation during the changeover from the injection phase to the containment sump recirculation phase during a design basis LOCA. Due to its importance in removing decay heat during shutdown conditions, system operation under reactor coolant system reduced inventory and mid-loop conditions was also reviewed. The inspection procedure used for this effort was IP 71111, Attachment 21.

.1 System Design - Residual Heat Removal System Mechanical, Electrical, and Instrumentation and Controls

a. Inspection Scope

The team reviewed design and licensing basis documents for the RHR system to determine the system and component functional requirements during normal operation and accident mitigation. The design and licensing documents reviewed for the RHR system included the Updated Final Safety Analysis Report (UFSAR), the plant Technical Specifications (TS), and the design basis document. In addition, the team reviewed component vendor manuals, engineering analyses and calculations, equipment qualification records, instrument setpoints, plant procedures, plant modifications, piping and instrument drawings, electrical schematics, instrumentation and control drawings and logic diagrams. The team also reviewed selected portions of design documents for interfacing systems such as the safety injection and containment building spray systems. For these systems, the team assessed the capability of the supporting systems to satisfy the design functions of the RHR system.

The team selected several major components for in-depth inspection. The components included the RHR pumps with their associated equipment, such as the seal coolers, the refueling water storage tank (RWST) with its level instrumentation, and several motor-operated valves. The team reviewed this equipment to assure adequate control of reactor coolant system temperature and water level during reduced inventory and mid-loop operations and the adequacy of a recent design change that increased the time available to operators for transferring from the injection phase to the sump recirculation phase during a design basis LOCA.

For selected calculations and analysis, the team reviewed the design basis functional requirements and assumptions to verify that they were appropriate and agreed with the current plant configuration, that proper engineering methods and models were used, and that there were adequate technical bases to support the conclusions. When appropriate, the team performed independent calculations to evaluate the adequacy of the document. Additionally, plant procedures including emergency operating procedures (EOPs) and surveillance tests were reviewed to ensure they supported the RHR system licensing and design basis. The team reviewed the licensee's evaluations of eight NRC Information Notices that pertained to RHR system operation or components.

In reviewing modifications, the team assessed the ability of the RHR system to perform its design functions, assuring that the changes did not adversely affect its operation. The team verified the adequacy of supporting engineering documents and post modification testing for selected modifications. During plant walkdowns, the team observed the material condition of the RHR system and reviewed the installation of the associated electric power supply, instrumentation and controls to verify that the system was configured consistent with the design drawings.

b. Findings

The team identified two examples where incorrect and non-conservative design inputs were used in engineering calculations.

Calculation C-S-1-E-0130

Incorrect and non-conservative design inputs for RWST water level were used in a calculation for emergency core cooling system (ECCS) pump performance during design basis accident conditions. The ECCS pump flow rates predicted by the calculation were used in other calculations that supported a plant design change and proposed emergency operating procedure (EOP) revision. Design change DCR-00-013, "Manual Transfer to Cold Leg Recirculation," was issued to change the plant design basis which would give operators more time to perform the manual actions required to changeover from the injection mode to the sump recirculation mode of ECCS operation without developing a vortex in the pump suction lines from the RWST. The increased response time allowance formed the basis for a proposed revision of EOP ES-1.3, "Transfer to Cold Leg Recirculation."

The licensee identified that some operating crews were unable to meet the design basis ECCS transfer time of 105 seconds. DCR-00-013 increased the allowable time to 180 seconds by lowering the assumed full ECCS flow/RWST drawdown rate and lowering the setpoint of the RWST empty alarm to 61.7 inches. The new alarm setting and drawdown flow rates and times were determined in calculations 4.3.05.31F, "RWST Vortex Study," and C-S-1-E-0130, "RWST Time to Vortex." To be consistent with the updated design information in DCR-00-013, new procedures for training all operating crews on the plant simulator were being developed. The licensee expected to begin qualifying all crews to these new procedures in April 2001. The new RWST empty alarm setpoint had not yet been changed.

The new ECCS pump flow rate was taken from Revision 2 of calculation SBC-535, “Seabrook ECCS Pump Performance for the Full Safeguards Condition.” A design input to the calculation was the RWST level at a tank empty alarm setting of 81.1 inches which the team concluded to be incorrect. For tank drawdown rate purposes, the low-low RWST level alarm setting of 116.6 inches is more appropriate than the empty alarm setting since it provides an additional 3 feet of pump suction pressure. Higher pump flow rates could reduce the operator response time estimated in DCR-00-013.

The team concluded that the error would likely be inconsequential since: (1) the expected change in pump flow rate would not be large; (2) tank drawdown calculations rounded up the SBC-535 predicted flow rate by about 30 gallons per minute; and (3) the estimated operator response time of 3 minutes and 22 seconds was rounded down to 3 minutes in procedure EOP ES-1.3. The team’s conclusion was confirmed by a licensee evaluation which determined that the additional 3 feet of pump suction pressure caused the total delivered flow from both RHR pumps to increase by only 17 gallons per minute. The evaluation results demonstrated that the stated times and conclusions of calculation C-S-1-0130 were unchanged. However, use of incorrect design inputs to ECCS performance calculations, if left uncorrected, could become a more significant concern and adversely affect EOPs and core cooling capability. Errors also could cascade into subsequent calculations. Thus this finding had a credible impact on safety.

The team evaluated the finding using phase one of the NRC’s significance determination process as a mitigating system design and qualification deficiency that does not affect RHR system operability per Generic Letter 91-18 (Revision 1). As discussed above, compensating margins existed and, when applied to the calculation results, provided assurance of system functionality. Thus, the finding was determined to be of very low safety significance.

Calculation 4.3.5.30F

The team identified a non-conservative design input error in calculation 4.3.5.30F, “CBS System Setpoints”. When determining the Technical Specification (TS) required RWST volume alarm setpoint for level transmitter LT-2381, the licensee did not include an instrument loop uncertainty allowance.

TS 3.1.2.6.b(1) requires that a minimum of 477,00 gallons be stored in the RWST tank during Modes 1, 2, 3, and 4. The licensee uses level transmitter LT-2381 to assure that this inventory is maintained. While reviewing calculation 4.3.5.30F, the team determined that the TS contained volume alarm setpoint for this transmitter excluded an instrument loop uncertainty allowance of 1.29 inches. This level difference corresponds to 1577 gallons. The team noted that, with the calibrated alarm setpoint exactly at the TS value of 477,000 gallons, and with instrument loop uncertainty biasing the setpoint non-conservatively downward, the instrument could have generated an alarm below the required TS value. The licensee considered an allowance for the 1.29 inches difference was accounted for in a Shutoff Allowance of 16.5 inches when establishing the RWST empty alarm setpoint. However, after further consideration regarding the use of this method for allowance of instrument loop uncertainty, the licensee issued CR 01-02722.

The team considered that the effect of not including instrument loop uncertainty allowance for the RWST contained volume alarm setpoint would likely be inconsequential since: (1) the RWST inventory is typically 3,000 gallons above the required 477,000 volume; (2) plant operators verify the RWST required inventory every eight hours to assure TS requirements are met; and (3) a separate contained volume approach alarm, which is set to alarm 1 inch above the contained volume alarm, provides added assurance for maintaining the required inventory of 477,000 gallons. However, the use of non-conservative inputs to ECCS design and performance calculations, if left uncorrected, could become a more significant concern and adversely affect EOPs and core cooling capability. Errors also could cascade into subsequent calculations. Thus this finding had a credible impact on safety.

The team evaluated the finding using phase one of the NRC's significance determination process as a mitigating system design and qualification deficiency that does not affect RHR system operability per Generic Letter 91-18 (Revision 1). As discussed above, compensating margins existed and provided assurance that the adequate RWST inventory would be available for all design basis conditions. Thus, the finding was determined to be of very low safety significance.

The use of incorrect and non-conservative inputs in design calculations was a violation of Criterion III, "Design Control," of 10 CFR 50, Appendix B. The team evaluated the conditions using phase 1 of the significance determination process and found them to be of very low safety significance (Green). The issues associated with this violation were entered into the licensee's corrective action program as CRs 01-02709 and 01-02722. This violation is being treated as a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy, issued on May 1, 2000 (65FR25368). (**NCV 05000447/2000-04-01**)

.2 Residual Heat Removal System Operation and Testing

a. Inspection Scope

The team reviewed selected operating and surveillance procedures and test results to verify that the RHR system was being operated, maintained, and tested in accordance with design and licensing requirements. Work orders, system health reports, and corrective actions taken to upgrade the RHR equipment, such as pump seals, and system valves and control components, were reviewed. The team reviewed the results of past containment closeout inspections regarding the condition of the containment sump screens. The adequacy of surveillance testing to ensure that 4213 gallons per minute would be delivered to the reactor coolant system during worst case accident conditions, as required by TS 4.5.2.h(3), also was reviewed.

The team reviewed the adequacy of the licensee's implementation of the Seabrook inservice test (IST) program for pumps and valves in the RHR and supporting systems. The review included applicable surveillance test procedures and focused on the ability of the RHR system to adequately cool down the plant and provide emergency cooling to the core during design basis accident conditions. Acceptance criteria included in the pump tests to satisfy the licensing and design basis conditions were reviewed. The team examined IST results for the two RHR system pumps to verify the ability of each pump to develop a head of greater than or equal to 171 pounds per square inch differential while operating on recirculation (560 gallon per minute). The team reviewed measurements of differential pressure, flow, and vibration recorded during each test and verified acceptable performance and trends. The team witnessed operators placing the system into service for normal plant cool down and for the quarterly pump and valve TS surveillance test.

The team reviewed IST procedures and test results for selected motor-operated, air-operated, check, and relief valves with regard, as applicable, to actuator and valve type, normal, safety, and fail positions, system location, valve class, category, size, and test frequency. Justifications for extended test intervals also were reviewed. With regard to relief valves, the inspectors discussed the capabilities of the licensee's new hot and cold relief valve test facility. The facility is designed to comply with the thermal equilibrium and ambient temperature requirements of Section I 8.1.1 (d) and (e) of ASME OMa-1996, Appendix I. For RHR system check valves, the team reviewed the condition monitoring program for compliance with Appendix II of ASME OMa-1996 with respect to types of test, examination and preventive maintenance activities, and associated intervals.

b. Findings

No findings of significance were identified.

1RO2 Changes to License Conditions (IP 71111.02)

a. Inspection Scope

The team reviewed a sample of safety evaluations (SEs) performed by the licensee to verify that changes at the Seabrook station related to systems, structures, or components (SSCs) and procedures, as described in the Updated Final Safety Analysis Report (UFSAR), were reviewed and documented in accordance with 10 CFR 50.59. The SEs were selected from the changes performed during the past two years, taking into consideration the risk significance of the change and the impact on the three reactor safety cornerstones. The team also reviewed a sample of the safety reviews (SRs) or 10 CFR 50.59 screens associated with changes to SSCs and procedures for which the plant staff determined that a SE was not required. This review was performed to verify that the threshold for performing SEs was consistent with 10 CFR 50.59. The team's review included eight SEs and eleven SRs. Portions of other SEs and SRs were also evaluated while selecting the specific sample for review and as input to determine the effectiveness of the problem identification and resolution process in the 10 CFR 50.59 area. The team reviewed each of 58 problem condition reports (CRs) issued during the year 2000:

- To establish the character of the problems identified in the 10 CFR 50.59 area during that time period;
- To determine how the licensee evaluated the same information;
- To determine if corrective actions were in progress for the identified issues.

The specific SEs and SRs reviewed are listed in Attachment 1 to the inspection report.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Identification and Resolution of Problems (IP 71152)

a. Inspection Scope

The team reviewed the licensee's activities associated with the identification and resolution of problems associated with the RHR system. The team conducted a plant walkdown and reviewed work orders, plant modifications, operating experience reports, system reports, Quality Assurance audits and surveillance reports to assess the licensee's adequacy of identifying problems. The team reviewed a sample of condition reports (CRs) associated with the RHR system and 10 CFR 50.59 process to assess the scope of identified problems and to evaluate the adequacy and timeliness of the corrective actions resulting from the identified problems. The team also reviewed operability evaluations and verified the completion of corrective actions.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

The inspectors presented the results of the inspection to Messrs. G. St. Pierre and J. Vargas and other members of the licensee management and staff at the conclusion of the inspection on March 23, 2001. The licensee acknowledged the inspection findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. Where proprietary information was identified it was returned to the licensee after review.

ATTACHMENT 1- SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

North Atlantic Energy Service Corporation

S. Doody	RHR System Engineer
R. Faix	Mechanical Design Supervisor
J. Grillo	Assistant Station Director
R. Hickok	NRC Coordinator
G. Kotkowski	Electrical Design Supervisor
J. Malone	50.59 Review Coordinator
E. Metcalf	Plant Engineering Supervisor
T. Nichols	Plant Engineering Manager
J. Peschel	Licensing Manager
G. St. Pierre	Station Director
J. Vargas	Director of Engineering
R. White	Mechanical Engineering Manager

Nuclear Regulatory Commission (NRC)

J. Linville	Acting Deputy Director, DRS
J. Brand	Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

05000443/2001-004-01	NCV	Inadequate Design Inputs in Engineering Calculations

ATTACHMENT 2 - LIST OF DOCUMENTS REVIEWED

Design Drawings

1-NHY-310002	Unit Electrical Distribution One Line Diagram
1-NHY-310087	Residual Heat Removal Pump 1-P-8B Control Schematic Diagram
ILD-1-CBS-Lo2381,	Instrument Loop Diagram Refueling Water Storage Tank TK-8 Level 1-CBS-L2381
1-NHY-310900,	Containment Sump Isolation Valve - V8 Schematic Diagram
PID-1-CBS-B20233	Containment Spray System
PID-1-CC-D20207	Primary Component Cooling Loop A Detail
PID-1-RC-D20841	Reactor Coolant System Loop No. 1
PID-1-RH-B20660	Residual Heat Removal System Overview
PID-1-RH-B20662	Residual Heat Removal System Train A Detail
PID-1-RH-B20663	Residual Heat Removal System Train B Cross-Tie Detail
PID-1-SI-B20450	Safety Injection System Low Head Injection (Accumulators) Detail
PID-1-SI-B20446	Safety Injection System Intermediate Head Injection System Detail
PID-1-SI-B20447	Safety Injection System High Head Injection System Detail
PID-1-SI-B20448	Safety Injection System Low Head Injection System Detail Sheet 1
PID-1-SI-B20449	Safety Injection System Low Head Injection System Detail Sheet 2
PID-1-SI-D20450	Safety Injection System Low Head Injection (Accumulators) Detail
Pittsburgh DesMoines Drawing 8(14084)	Refueling Water Storage Tank Mixing Chamber Details
Pittsburgh DesMoines Drawing E6(14084)	Refueling Water Storage Tank Mixing Chamber Erection
Joseph Oats Drawing No. 5620	Vertical RHRHX Assembly & Details
Joseph Oats Drawing No. 5628	Vertical RHRHX Details
Joseph Oats Drawing No. 5627	Vertical RHRHX Details
Joseph Oats Drawing No. 5622	Vertical RHRHX Details
PSNH Drawing No. 9763 1-CBS-1211-01	Sleeve
PSNH Drawing No. 9763 1-CBS-1211-02	Containment Spray
PSNH Drawing No. 9763 FI-172-01	Encapsulation Tank Field Instruction
Westinghouse Drawing. No. 9504D34	RHRP AC Motor Outline
PXE Drawing No. 578	Sump Isolation Valve Encapsulation Section & Details and General Arrangement

ATTACHMENT 3 - LIST OF DOCUMENTS REVIEWED (cont'd)

Design Bases Documents

DBD-RH-01	Residual Heat Removal System
UFSAR 5.4.7	Residual Heat Removal System
UFSAR 6.2.2	Containment Heat Removal System
UFSAR 6.3	Emergency Core Cooling Systems

Engineering Calculations

C-S-1-83801,Rev. 0	RHR Pump NPSH During Injection and Recirculation
C-S-1-83804, Rev. 0	Allowable Leakage from Containment Recirculation Sump Isolation Valve Encapsulation
C-S-1-84011, Rev. 0	Plant Specific Values for OS1201.10, Shutdown LOCA
C-S-1-E-0130, Rev. 1	RWST Time to Vortex
4.3.05.10.F, Rev. 6	CBS Hydraulic Analysis
4.3.05.30.F, Rev. 8	CBS System Setpoints
4.3.05.31.F, Rev. 3	RWST Vortex Studies
4.3.03.09.F, Rev. 2	Residual Heat Removal Thermal Relief Valves
SBC-535, Rev. 1	ECCS Pump Performance for Full Safeguards Condition
SBC-535, Rev. 2	ECCS Pump Performance for Full Safeguards Condition
SBC-535, Rev. 3	ECCS Pump Performance for Full Safeguards Condition
SBC-535, Rev. 4	ECCS Pump Performance for Full Safeguards Condition
SBC-727, Rev. 0	RHRS Overpressure Protection
9763-3-ED-00-83-F, Rev. 6	Emergency Diesel Generator (EDG) Loading
9763-3-ED-00-66-F, Rev.3	Control Circuit Voltage Drop Calculation.
9763-3-ED-00-14F, Rev.10	125 V dc System Battery, Charger, Motor Feeders
763-ED-00-23-F, Rev. 4	Medium Voltage Protective Relay Co-ordination and Miscellaneous Relay Settings

Engineering Evaluations and Design Changes

DCR 88-161	RHR Pump Redesign To Reduce Trust Loads
DCR 88-161	RHR Pump Redesign To Reduce Trust Loads
DCR 00-0013	Manual Transfer To Cold Leg Recirculation Timing
DCR 00-0010	Reactor Vessel Narrow Range Level Instrumentation Modifications
MMOD 99-611	Replacement of RHR Flow Switches
EE 90-27	Evaluation of Full Flow Testing for RHR Pumps

EE 92-027	Evaluation of PCCW Temperature Transient During Post-LOCA Recirculation
EE 95-01	Evaluation of Operation with Steam Generator Nozzle Dams
EE 95-10	Evaluation of Operation With Steam Generator Nozzle Dams,
EE-99014 Rev. 0	Transfer to Cold Leg Recirculation Timing Criteria
EE-99014 Rev. 1	Transfer to Cold Leg Recirculation Timing Criteria
EE-00013, Rev. 0	Transfer to Recirculation Mode/Cooling Tower Operation - EOP Task Analysis
SS-EV-980006 Rev.1	IST Pump Surveillance Requirements
NSAL 95-003-	Potential to Exceed 110% of RHRS Design Pressure

<u>10CFR 50.59 Safety Evaluations (SE)</u>

BCR No. 00-01	SE dated 1/12/00 for Technical Specification 3/4.7.6 Bases Change
DCR 98-044.	SE dated 6/2/99, for DBD-RH-01 Revision for RHR System
DCR 99-0020	SE dated 11/24/99 for the RHR Suction Isolation Valve Interlock
DCR 99-0036	SE dated 5/10/00 for the Addition of Pressure Gages for Comprehensive IST Pump Tests
DCR 00-0016	SE dated 10/4/00 for the Control Building Air (CBA) Emergency Clean-up Filter Temperature Switches
DCR 99-007	SE dated 7/7/99 for the Service Water System Vacuum Breaker Configuration Change
DCR 99-0015	SE dated 8/18/99 on Nuclear Instrumentation (NI) Start Up Rate Meter Replacement
DCR 99-0031	SE dated 2/16/00 for the OR07 Service Water Piping Refurbishment
OS1013.03 Rev. 09	SE for the Procedure Revision for RHR System Flush
UFCR 00-002	SE dated 2/2/00 for the Engineered Safety Features Actuation System (ESFAS) Slave Relay Testing, UFSAR Change
UFCR 99-048	SE dated 11/10/99 for the Accident Monitoring Instrumentation (AMI) Power Source Requirements Change, UFSAR Change.

ATTACHMENT 3 - LIST OF DOCUMENTS REVIEWED (Cont'd)

<u>10CFR 50.59 Nuclear Safety Screen Evaluations</u>

TMod 00-0020	1-CL-5611-01-L1-4 Leak Repairs
Tmod 00-0008	Steam Blowdown Coating Repair
Tmod 00-0015	Temporary Power for Metal Detectors
MMod 99-0524	RHR Pump Oil Sample Collection Modification
MMod 99-0611	RHR Pump Flow Alarm Setpoints
MMod 99-0597	Yarway 5600 Valve Substitution
MMod 98-0658	Equipment Qualification Required Maintenance for Valcor Solenoid Valves.
MMod 00-0543	Containment Air Purge System.
Procedure LS0557.19	Changes on 480 VAC Breaker Refurbishment
Procedure MS0517.13	Change for Concrete Grout.
Procedure OS1023.51	Control Room Ventilation & Air conditioning Changes

<u>Condition Reports</u>

ACR 99-4525	"Operability Determination - Non-Condensable Gas in 'B' RHR Pump Suction Piping," dated November 9, 1999
ACR 99-2038	Cold Leg Recirculation Manual Operator Time Not Met
ACR 99-2929	RH-P-8B IST Results
ACR 99-4193	Operators Exceeding 108-Second Criterion in ES-1.3
ACR 99-4557	RHR Pump Quarterly Vibration Test Results
ACR 99-4606	RH-P-8B Oil Sample Test Results
CR 97-03772	Relief Valve Design Requirement Discrepancies
CR 98-03832	DCR 96-008 RWST Setpoint Change May Not Have Accounted for Impact to Operator Response Times
CR 99-04608	EOP Task Analysis Deficiency
CR 99-06908	Inadequate Review of UFSAR during Development of EOP
CR 99-07057	Encapsulation Tank Leakage
CR 00-00491	Leakage Past Test Header Isolation and Relief Valves
CR 00-00734	Leak Identified on RH-P-8A from Mechanical Seal
CR 00-03025	RH-P-8B Pump Exceeds Vibration Limits
CR 00-04768	Previously Identified Condition Determined to be Safety Significance Under New Revision to the SSOE
CR 00-09151	Heat Exchanger Thermal Performance Program
CR 00-10062	Review of SEN 214

CR 00-10564	Highest RCS Temperature to Realign RHR Pump to RWST
CR 00-10101	Error Found in Engineering Evaluation EE-99014
CR 00-11553	Train A Exceeded Maximum Pump Flow During Cold Shutdown Valve Testing
CR 00-12420	Procedure MS0523.24, Specified Incorrect Lubricant
CR 00-12808	Results of RH-P-8A Comprehensive Inservice Pump Test
CR 00-13092	Upper Bearing Installation Tool Caused Damage
CR 01-00747	RHR Flow Loop Controller Sections Not Calibrated Since Mid-1980's
CR 01-01114	Containment Sump Isolation Valve Position Indication Should Be Type A Variable
CR 01-02709	Calculation SBC-535 Used Incorrect RWST level
CR 01-00193	Event and Root Cause Report for RC-V89 RHR Suction Relief Valve Lift, dated January 29, 2001

<u>Station Procedures</u>

ES1804.056, Rev. 02	CBS-TK-101A and CBS-TK-101B Encapsulation Tank Leakage Rate Test
ES1807.021, Rev. 00	Level I Vibration Trending and Analysis
ES1850.011, Rev. 02	Relief/Safety Valve Testing Program
EX1804.044, Rev. 05	Relief Valve Setpoint Pressure And Leakage Test
EX1850.015, Rev. 00	Check Valve Condition Monitoring Program
IX1622.225, Rev. 04	L-2383 Refueling Water Storage Tank Level Instrumentation Calibration
IX1622.224, Rev. 05	L-2380 Refueling Water Storage Tank Level Instrumentation Calibration
IX1622.232, Rev. 05	L-931 Refueling Water Storage Tank Level Calibration
IX1622.225, Rev. 05	L-2383 Refueling Water Storage Tank Level Calibration
OX1426.20, Rev. 02	Diesel Generator 1A 18 Month Operability and Engineered Safeguards Pump and Valve Response Time Testing Surveillance
OX1426.21, Rev. 01	Diesel Generator 1B 18 Month Operability and Engineered Safeguards Pump and Valve Response Time Testing Surveillance
MS0523.24, Rev. 06	Ingersol-Rand RHR Pump Maintenance
OP 9.2, Rev. 09	Emergency Operating Procedure User's Guide
OS1000.01, Rev. 11	Heatup From Cold Shutdown to Hot Standby
OS1000.12, Rev. 03	Operation With RCS At Reduced Inventory/Midloop Conditions

OS1013.03, Rev. 10	Residual Heat Removal Train A Startup And Operation
OS1013.04, Rev. 10	Residual Heat Removal Train B Startup and Operation
OS1013.05, Rev. 08	Residual Heat Removal Train A Shutdown
OX1406.13, Rev. 05	Containment Recirculation Sump Valve Cold Shutdown Test
OX1406.12, Rev. 05	18 Month Containment and Containment Spray Recirculation Sump Surveillance
OX1413.01, Rev. 09	A Train RHR Quarterly Flow and Valve Stroke Test and 18 Month Valve Stroke Observation
OX1413.05, Rev. 04	RHR Cold Shutdown Valve Testing
OX1413.06, Rev. 02	RHR/RC Suction Valve 18 Month Interlock Verification Surveillance
OX1413.07, Rev. 00	RH-P-8A Comprehensive Pump Test
OX1413.08, Rev. 00	RH-P-8B Comprehensive Pump Test
OX1456.19, Rev. 06	Post-Accident Monitoring Monthly Channel Checks
OX1408.04, Rev. 06	Weekly Borated Water Source Evaluation
OS1001.11, Rev. 01	Reactor Coolant System Shutdown Level Instrumentation
OS1000.14, Rev. 03	Reactor Coolant System Evacuation and Fill
IN1662.230, Rev. 2	L-9465 RCS Loop Ultrasonic Level Instrument Calibration.
IN1662.231, Rev. 2	L-9466 RCS Loop Ultrasonic Level Instrument Calibration.
RM 6.1, Rev. 03	Receipt, Processing, and Approval of Vendor Documentation
STP-105, Rev. 00	RHR/Reactor Coolant Mid-Loop
EDI No. 30230	Engineering Department Instruction - Foreign Print System

Preoperational Tests

1-PT(I)-42.1, Rev. 1	RHR Heat Exchanger Performance Test Results, 1/21/86
1-PT-12.3, Rev. 0	RWST and SAT Drawdown Test, 4/14/86

NRC Information Notice Response

AR 98013184	Evaluation of IN 98-23: Crosby Relief Valve Setpoint Drift Problems Caused By Corrosion of the Guide Ring," dated October 13, 1998

AR 98001775	Self-Assessment of IN 97-60: Incorrect Unreviewed Safety Question Determination Related to Emergency Core Cooling System Swapover from the Injection Mode to the Recirculation Mode," dated June 29, 1999
AR 97015993	Response to IN 97-40: Nitrogen Accumulation from Backleakage from SI Tanks," dated May 14, 1998
AR 97015936	Evaluation of IN 97-33: Unanticipated Effect of Ventilation System on Tank Level Indications and Engineering Features Actuation System Setpoint," dated November 10, 1998
AR 98020029	Evaluation of IN 98-40: Design Deficiencies Can Lead To Reduced ECCS Pump Net Positive Suction Head During Design-Basis Accidents," dated June 22, 1999
AR 98002284	Evaluation of IN 98-02: Nuclear Power Plant Cold Weather Problems and Protective Measures," dated January 11, 1999
CR 00-04771	Evaluation of IN 2000-08: Inadequate Assessment of the Effect of Differential Temperatures On Safety-Related Pumps," dated August 6, 2000
OE12308	IN 95-08 - Inaccurate Data Obtained with Clamp-on Ultrasonic Flow Measurement Instruments," dated February 14, 1995

<u>Other</u>

Electrical EQ File No. 113-22-01, Rev. 2	600V Control Cables
Electrical EQ File No. 248-47-01, Rev. 2	Conax P/N 7873-10000
Vendor Manual W120-6	Aux. Heat Exchanger Instructional Manual
Vendor Manual I075-20	RHR Pump Installation and Maintenance Manual
NAH-3092, Rev. 1	Westinghouse Letter, Post-LOCA RHR Valve Alignment, dated May 29, 1986
NAH-U-1759	Westinghouse Letter, RWST Level Instruments and Switchover Sequence," dated July 31, 1979
NYN-98001	30-Day Response to Generic Letter 97-04, dated 1/5/98
Engineering Work Request 97-095	UFSAR/SITR Discrepancies," dated September 17, 1999
Engineering Work Request 00-0067	Manual Transfer to Cold Leg Recirculation (108 Second Limitation), dated June 21, 2000
SORC Review Comments Summary	"Manual Transfer to Cold Leg Recirculation Timing," dated January 29, 2001
SORC Minutes	Meeting 01-008, dated February 8, 2001

ATTACHMENT 3 - LIST OF DOCUMENTS REVIEWED (Cont'd)

Memorandum Abdelghany to Vargas	"Seabrook ECCS Pump Runout and ECCS System Technical Specification Changes - Cycle 5 " Group LOCA-SB 93-009, SBP93-0443, dated September 15, 1993
Instructor Guide	RHR/Midloop Operations, dated August 18, 2000
Walkdown Report for OR07	RHR System
Operations Standing Order No. 01-004	"Upgrade of CBS-V8 & 14 Position Indication To Accident Monitoring Instrumentation," dated February 5, 2001

Attachment 3 - LIST OF ACRONYMS USED

ASME	American Society of Mechanical Engineers
CBS	Containment Building Spray
CFR	Code of Federal Regulations
CR	Condition Report
DCR	Design Coordination Report
ECCS	Emergency Core Cooling System
EOP	Emergency Operating Procedure
IST	In-service Testing
LOCA	Loss-of-Coolant Accident
MOV	Motor-Operated Valve
NRC	Nuclear Regulatory Commission
NAESCO	North Atlantic Energy Service Corporation
RHR	Residual Heat Removal
RWST	Refueling Water Storage Tank
SE	Safety Evaluation
SSC	System, Structures, or Components
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report