October 5, 2000

Mr. Harold W. Keiser
President and Chief Nuclear Officer
PSEG Nuclear LLC - X04
P. O. Box 236
Hancocks Bridge, NJ 08038

Subject: HOPE CREEK NRC INSPECTION REPORT 05000354/2000-009

Dear Mr. Keiser:

On August 25, 2000, the NRC completed a team inspection of the service water and safety auxiliaries cooling water systems and the evaluation of changes, tests, and experiments at your Hope Creek facility. The enclosed report presents the results of that inspection. The preliminary findings were discussed with Mr. D. Garchow on August 25, 2000, and in several subsequent telephone conversations concluding on September 7, 2000.

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.

No significant problems were identified in the service water and safety auxiliary cooling water systems. However, the team identified three issues of very low safety significance (Green) that you have entered into your corrective action program and are discussed in the summary of findings and in the body of the attached inspection report. One of these issues indicated some engineering weaknesses which included multiple examples of inadequate design control concerning incorrect information in calculations and drawings of record. The other two issues included failures to test and calibrate some system instruments as required and to perform an adequate review of operating experience notifications concerning ultrasonic flow meters. The three issues were determined to be violations of NRC requirements, but because of their very low safety significance the violations are not cited. If you contest these non-cited violations, 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 Hope Creek Generating Station.

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Sincerely,

/RA/

Wayne D. Lanning, Director
Division of Reactor Safety

Docket No. 05000354
License No. NPF-57

Enclosure: NRC Inspection Report No. 05000354/2000-009

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REGION I

Docket/Report No:	05000354/2000-009
License No:	NPF-57
Licensee:	Public Service Electric and Gas Company (PSEG)
Facility:	Hope Creek Nuclear Generating Station
Location:	P.O. Box 236 Hancocks Bridge, NJ 08038
Dates:	August 7-11 and 21-25, 2000 August 28-31 and September 7, 2000 (Region I - In-office)
Inspectors:	L. Prividy, Senior Reactor Inspector, Team Leader, DRS S. Chaudhary, Senior Reactor Inspector, DRS G. Morris, Reactor Engineer, DRS T. Burns, Reactor Engineer, DRS J. Carrasco, Reactor Engineer, DRS
Accompanied by:	Lisa Mitchell, Region I Summer Intern Fatima Salaam, Region I Summer Intern
Approved by:	Lawrence T. Doerflein, Chief Systems Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 050000354-00-09; on 8/7-11, 21-25, 28-31&9/7/2000; Public Service Electric and Gas Company; Hope Creek Generating Station; Mitigating Systems; Other activities (PI&R).

The inspection was conducted by a region-based team of the station service water (SSW) and the safety auxiliaries cooling systems (SACS) 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 (see Attachment 2).

Cornerstone: Mitigating Systems

- GREEN. The team found that the licensee had failed to control the inputs and assumptions used in the calculations for determining motor operated valve (MOV) thermal overload (TOL) protection, the accuracy of the SSW system ultrasonic flow meters, the loop accuracy of the SACS temperature instrumentation, and the leakage past the seat of two SACS isolation valves associated with the residual heat removal heat exchangers. Also, the team found that the licensee had failed to provide adequate design review of the logic diagrams and P&ID drawings for the SSW and SACS systems. These failures to correctly provide adequate design inputs and assumptions in calculations and properly review design drawings were determined to be of very low risk significance (GREEN) by the SDP phase 1 screening. This conclusion was primarily based on the MOV TOL issue which was a programmatic problem with credible potential to impact safety and was more than isolated. While the sizing and the settings of the TOLs could impact the functionality of several SSW system MOVs, no immediate impact to the operability of the system was apparent due to compensating margins and testing. The examples noted above were considered to be a non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control", and were included in the licensee's corrective action program. (Section 1R21.1, Design-Mechanical, Electrical and Instrumentation and Control)

- GREEN. The team found that the licensee had missed their prescribed calibration of the ultrasonic flow meter instruments used during the SSW pump in-service tests and the temperature transmitter/temperature switch for the SACS heat exchanger bypass isolation valve. These failures to test and calibrate these instruments as required to support testing and demonstrate their operational readiness were determined to be of very low risk significance (GREEN) by the SDP phase 1 screening. This conclusion was primarily based on the SSW flow meter issue which was a programmatic problem with credible potential to impact safety and was more than isolated. The operability of the "A" SSW pump was questioned during the inspection due, in part, to the operation of the SSW flow meters. However, this operability question was ultimately resolved after rezeroing the flow meters and achieving a satisfactory retest. The lack of testing and calibrating these instruments was a non-cited violation of 10 CFR 50, Appendix B, Criterion XI, “Test Control.” The issues associated with this violation are in the licensee’s corrective action program. (Section 1R21.2 Operations, Maintenance, Surveillance, and Testing)

Cornerstone: Cross-Cutting

- GREEN. The team identified that the licensee missed two opportunities to question the use of the ultrasonic flow meters in the SSW system through their Operating Experience program. The failure to provide adequate review of operating experience notifications was determined to be of very low risk significance by the SDP phase 1 screening. The examples suggested a programmatic problem concerning operating experience information with credible potential to impact safety and was more than isolated. There was no immediate impact to the operability of the system based on the satisfactory retest of the “A” SSW pump. The licensee’s failure to adequately assess the operating experience notifications after they had been entered into their corrective action system was an example of a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action.” The issues associated with this violation are in the licensee’s corrective action program. (Section 4OA1, Identification and Resolution of Problems)

Report Details

1. REACTOR SAFETY
Cornerstone: Mitigating Systems

1R21 Safety System Design and Performance Capability

Introduction

The station service water (SSW) and the safety auxiliaries cooling system (SACS) at the Hope Creek Generating Station (HCGS) were reviewed using Procedure 71111, Attachment 21. The SSW and SACS were selected because they are risk significant mitigating systems which provide cooling water to equipment required for safe reactor shutdown.

.1 Design - Mechanical, Electrical and Instrumentation and Controls

a. Inspection Scope

The team reviewed the SSW and SACS system design and licensing basis documents to determine the system functional requirements during abnormal and accident conditions. For the documents reviewed, which included calculations and analyses, the team verified that the assumptions were appropriate, that proper engineering methods and models were used, and that there was an adequate technical basis to support the conclusions. Where possible, the team performed independent calculations to evaluate the document adequacy. The review was performed to verify that: (1) the design basis was in accordance with the licensing commitments and regulatory requirements; (2) the design output documents such as drawings and procurement specifications were correct; and (3) the installed system and components were tested to verify the design bases were met. The system designs were reviewed to verify that modifications that were performed satisfied the design intent.

The team reviewed the Updated Final Safety Analysis Report (UFSAR) to establish the design and licensing basis for the SSW and SACS and interfacing systems. The piping and instrumentation drawings (P&IDs), the configuration baseline documents and the installed configuration were also reviewed to assess the capability of the system to satisfy the design intent. The team performed walkdowns of selected system components such as pumps, traveling water screens, heat exchangers, strainers, control station locations, electrical power supplies, and motor control centers.

The team selected the SSW pumps and the SACS heat exchangers for a detailed review. In addition, the follow-up actions associated with a service water system operational performance inspection conducted in 1996 and 1997 by PSEG were reviewed.

The team also conducted other component design reviews, including components from supporting systems, such as:

- Valves used for isolation between safety related and non-safety related interfaces
- Bypass valves around heat exchangers
- Instrumentation and control including local and main control room control and remote safe shutdown control
- The design requirements for pump drivers and valve operators
- Electrical voltage relay setpoint for motor overcurrent protection and MOV thermal overload selection

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

The team found that the HCGS electrical protective design for the motor operated valves (MOVs) was inconsistent with the description in the FSAR Section 8.3.1.1.2.10, in that the selection of the thermal overload (TOL) protection for the MOVs was based on both the C1 and C2 positions from Regulatory Guide 1.106, “Thermal Overload protection for Electric Motors on Motor-Operated Valves.” The purpose of this regulatory guide was to ensure that either (1) the TOL was bypassed under accident conditions, or (2) the protective devices were selected with sufficient margin for the valve to perform its safety function. Contrary to the intent of the guide, the licensee in calculation E-018, Rev. 1, “Selection of Overload Heaters for AC Motors,” used both positions. They continuously bypassed the TOL trip function and also selected the setpoints too high to provide sufficient protection for the limited time duty motors provided for the valve operators. This condition could have resulted in undetected damage to the motor prior to any warning alarm. Furthermore, the team found that 6 of 8 inspected TOLs installed on SSW MOVs were even larger than those specified by calculation E-018. The team had no operability concerns since acceptable motor currents were being demonstrated during MOV testing. The licensee initiated Notifications 20038461 and 20038462 to address these concerns through their corrective action program.

The team reviewed the instrument inaccuracy calculation, H-1-EA-IST-2218, for the SSW header flow meters (1EAFIT-2218A/B) and compared the calculation assumptions and inputs to the as-installed device, the vendor’s technical manual and the instrument re-calibration frequency. The calculation assumed a base instrument equal to that obtained in the manufacturer’s calibration lab under controlled conditions and a re-calibration frequency of 18 months. The team found that the application of the flow meter at HCGS was different than that used in the laboratory calibration both in physical installation details and the characteristics of the flow medium. The team noted that the vendor’s technical manual indicated a number of factors that had the potential to affect instrument accuracy. These factors had not been addressed in the calculation. In fact, the licensee had recognized that changing silt and salt levels in the SSW had required multiple re-calibrations as late as July 10, 2000 (Notification 20034221), but had failed to relate that problem to the calculation. In addition, the calculation had not been updated to assess the potential effect on accuracy when the calibration frequency was changed from 18 months to 36 months in 1994. The licensee initiated Notifications 20038170 and 20038256 to address these concerns through their corrective action program.

The team reviewed the instrument inaccuracy calculation, SC-EG-0150-1, “Safety Auxiliaries Cooling System Heat Exchanger Outlet Temperature”, which justified the increase in the calibration period to 1217 days (40 Months). The team identified that the total loop accuracy calculation for both the temperature switch and the temperature indicators used the wrong instrument accuracy for the temperature element. The calculation of the total loop drift used 36 month drift data for the temperature element and failed to use the previously calculated 40 month drift value for the temperature transmitter. The calculation of the total loop allowance, used to calculate the margin from the allowable value, failed to address the process measurement accuracy term in the equation for margin. Also, a prior calculation for these instruments, CS-117, “SACS Heat Exchanger Temperature {Switches}”, which was issued in 1985 for an 18-month calibration period, had never been superseded. Notification 22038803 was initiated to address this concern on calculation discrepancies.

The team found a design input error in a vendor calculation supporting safety evaluation H000-010, which evaluated the operability of the SACS return isolation valve (EG-HV-2512A) for the RHR heat exchanger without its normally installed rubber seat. The licensee had supplied the vendor a differential pressure of 112 psid to calculate the maximum expected seat leakage without the rubber seat. Subsequently, the licensee learned that MOV calculation H-1-EG-MDC-0938 specified a maximum possible DP for this valve of 150 psid. There was no impact on operability since the revised valve leak rate using the higher DP was still substantially less than the maximum allowable leak rate of 3000 gpm. The licensee issued notification 20037546 to document this discrepancy and take appropriate corrective action.

During the course of the inspection, the team identified multiple deficiencies in logic drawing errors, P&ID control channel assignment errors and in the configuration baseline document describing the SACS pump automatic starting design feature. In response to these concerns, the licensee issued Notifications 20038463, 20038392 and 20038339.

The failures to correctly provide adequate design inputs and assumptions in the above calculations and properly review design drawings were determined to be of very low risk significance (GREEN) by the SDP phase 1 screening. This conclusion was primarily based on the MOV TOL issue which was a programmatic problem with credible potential to impact safety and was more than isolated. While the sizing and settings of the TOLs could impact the functionality of several SSW system MOVs, no immediate impact to the operability of the system was apparent due to compensating margins and testing. The examples noted above were considered to be a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control. The issues associated with this violation are in the corrective action program as listed above. **(NCV 05000354/2000-009-01)**

As part of the request for applicable calculations for the SSW system, the team had requested a number of instrumentation calculations, including calculation SC-EA-0508, Loop Tolerance Calculation for 1EA-PSH-2014A, B. This instrument provides an input to the common SSW System Trouble alarm. This calculation was not available for review by the team. The licensee had identified that this calculation had been missing (CR 960606248) and it had been scheduled to be redone by September 15, 1998.

Although this problem was in the corrective action program, the calculation had not been reconstructed.

.2 Operations, Maintenance, Surveillance and Testing

a. Inspection Scope

The team reviewed selected operating and preventive maintenance procedures to assess whether the selected systems were being operated and maintained in accordance with the system requirements. Work orders, system health reports, and various corrective actions taken to upgrade the SSW pumps and traveling water screens were reviewed. The adequacy of surveillance testing to ensure that adequate cooling water flow would be supplied to the safety related components during worst case accident conditions was also reviewed. The team reviewed the test procedures and data for the four SSW pumps for the past two years. A surveillance test and retest of the “A” SSW pump was witnessed.

b. Issues and Findings

The team reviewed the calibration of SSW flow meters 1EAFIT-2218A and B to determine their acceptability to support in-service testing (IST) of the SSW pumps. The team learned that the prime standard calibration frequency for these ultrasonic flow meters had been specified by the licensee as six years, but they had never been prime standard calibrated since installation in 1992. The instruments are currently scheduled for calibration in October 2000. The licensee issued Notification 20037748 to address the concern of the missed calibration. In addition to this oversight and consistent with the discussion in the previous report section regarding these instruments, the team observed that I&C engineering had missed prior opportunities to identify calibration discrepancies and assist in earlier resolution of SSW testing problems, especially between 1995-1997 when extensive calculation work was being done regarding SSW and SACS system performance under accident conditions. This team observation was supported during the inspection when the “A” SSW pump was declared inoperable after failing its IST acceptance criteria. However, subsequent interaction with the ultrasonic flow instrument vendor suggested that 1EAFIT-2218A should have been properly aligned and re-zeroed prior to each pump test. This change was made and the “A” SSW pump was satisfactorily retested and declared operable.

In addition, the team found the last calibrations for temperature transmitter 1EGTT-2535B1 and temperature switch 1EGTSH-2535B1 had been performed in 1996 and had not been rescheduled for re-calibration. The HCGS preventive maintenance program identifies that these calibrations should have been performed on a 36-month cycle. The licensee issued Notification 20038468 to address the concern of the missed calibration.

The failures to test and calibrate these instruments as required to support testing and demonstrate their operational readiness were determined to be of very low risk significance (GREEN) by the SDP phase 1 screening. This conclusion was primarily based on the SSW flow meter issue which was a programmatic problem with credible potential to impact safety and was more than isolated. The licensee declared the “A” SSW pump inoperable during the inspection when it failed to meet the IST acceptance criteria due, in part, to the operation of the SSW flow meters. However, this operability problem was ultimately resolved after rezeroing the flow meters and achieving a satisfactory retest. The lack of testing and calibrating these instruments was a non-cited violation of 10 CFR 50, Appendix B, Criterion XI, Test Control. The issues associated with this violation were in the corrective action program as listed above. **(NCV 05000354/2000-009-02)**

1RO2 Evaluations of Changes, Tests, and Experiments

a. Inspection Scope

The team reviewed Procedure No. NC.NA-AS.ZZ-0059(Q), Revision 3, “10CFR50.59 Program Guidance,” and Procedure No. NC.NA-AP.ZZ-0059(Q), Revision 7, “10CFR Applicability Reviews and Safety Evaluations.”

The team reviewed selected 10CFR50.59 safety evaluations (SE) representing the three cornerstones: Initiating events, mitigating systems and barrier integrity. The objective of this review was to verify that changes made to the facility or procedures as described in the UFSAR were reviewed and documented in accordance with 10CFR50.59, and the safety issues pertinent to the changes were properly resolved.

The team interviewed engineering personnel engaged in the preparation and the review of the selected 10CFR50.59 safety evaluations. Throughout the reviews of the selected SEs, the team conducted meetings with the licensee to resolve questions and observations made during the course of the review. The following 10CFR50.59 safety evaluations were reviewed:

H1999-005 - Reactor Protection System Operation
H1999-014 - Oscillation Power Range Monitor for ECR 97-000533
H1999-058 - Replace Portion of Existing Gland Seal Drain Lines with Flexible Hose Sections for HPCI Pump Turbine
H2000-033 - Reactor Water Cleanup System Operation
H1999-020 - RHR Shutdown Cooling during a LOP & the Alternate Shutdown Method
H2000-040 - SSW Lube Water Tank Replacement
H1999-003 - Primary Containment Integrity Verification
H1999-026 - Required Flow Rates and Heat Loads
H2000-011 - Technical Specification Bases Change, Reactivity Control Shutdown Margin
H1999-054 - Battery Replacement

The team also reviewed 10 change items (e.g., procedure and UFSAR changes) that were screened out of the 10 CFR 50.59 evaluation process to verify that such screenings were appropriate.

b. Issues and Findings

There were no findings identified.

4. OTHER ACTIVITIES (OA)

4OA1 Identification and Resolution of Problems (IP 71152)

a. Inspection Scope

For the station service water (SSW) and safety auxiliaries cooling system (SACS), the team reviewed the activities for identifying, evaluating and correcting problems which could impact the cornerstone objectives.

b. Issues and Findings

The team reviewed the operating experience assessment performed by the HCGS staff concerning problems identified by industry and the NRC with the operation and calibration of ultrasonic flow instrumentation similar to those used in the SSW system. NRC Information Notice 95-08, "Inadequate Data Obtained with Clamp-on Ultrasonic Flow Measurement Instruments" addressed problems with the effect of physical piping arrangements affecting accuracy and excessive damping of the flow signal hiding flow measurement problems. The team found that the licensee had failed to properly address this information notice since the ultrasonic flow meters were permanently installed in the SSW systems by 1993. The licensee initiated Notification 2003871 to address this concern.

The team noted that the licensee had entered Operating Experience (OE)10435, Controlotron model 990 Flow Meters, into their corrective action program on November 23, 1999, by Notification 20013420. However, the licensee could not find any evaluation that had been performed for HCGS. The licensee reissued the OE by Notification 20038235 for HCGS.

The failure to provide adequate review of operating experience notifications was determined to be very low risk significance (GREEN) by the SDP phase 1 screening. The examples suggested a programmatic problem concerning operating experience information that was more than isolated and had credible potential to impact safety. There was no immediate impact to the operability of the system based on the satisfactory retest of the "A" SSW pump. The licensee's failure to adequately assess the operating experience notifications after they had been entered into their corrective action system is a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action. The issues associated with this violation are in the corrective action program as listed above. **(NCV 05000354/2000-009-03)**
Also, findings regarding the identification and resolutions of problems were identified

and are described in Sections 1R21.1 and 1R21.2 of this report.

4OA5 Management Meetings

PSEG representatives were informed of the purpose and scope of the inspection at an entrance meeting conducted on August 7, 2000. The team presented the preliminary inspection findings to Mr. D. Garchow and other members of PSEG management on August 25, 2000, who acknowledged the findings presented. Several subsequent telephone conversations were held with PSEG personnel to further discuss the inspection findings. The last conversation occurred on September 7, 2000. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Public Service Electric and Gas Company

D. Garchow	Vice President, Engineering
G. Salamon	Manager, Licensing
F. Sullivan	Director, Nuclear Plant Engineering
J. Nagle	Licensing Engineer
V. Fregonese	Manager, Mechanical Design
G. O'Connor	Manager, Electrical and I&C Design

New Jersey Department of Environmental Protection

R. Pinney	Nuclear Engineer

Nuclear Regulatory Commission (NRC)

L. Doerflein	Chief, DRS Systems Branch
J. Schoppy	Senior Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

05000354/2000-009-01	NCV	Inadequate design inputs and assumptions in engineering calculations
05000354/2000-009-02	NCV	Failures to test and calibrate plant instruments
05000354/2000-009-03	NCV	Inadequate reviews and corrective actions taken in response to operating experience information

LIST OF ACRONYMS USED

CFR	Code of Federal Regulations
CR	Condition Report
DP	Differential Pressure
gpm	gallons per minute
HCGS	Hope Creek Generating Station
IST	In-service Testing
MOV	Motor-Operated Valve
NRC	Nuclear Regulatory Commission
PSEG	Public Service Electric and Gas Company
psid	pounds per square inch differential
SACS	Safety Auxiliaries Cooling System
SE	Safety Evaluation
SSW	Station Service Water
TOL	Thermal Overload
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report

ATTACHMENT 1
LIST OF DOCUMENTS REVIEWED

Configuration Baseline Documents

DE-CB.EG-0054	Safety and Turbine Auxiliaries Cooling System
DE-CB.EA/EP/EQ-0052	Station Service Water System

Calculations

E-004.1(Q)	Rev.12	HC Class 1E 125 VDC Battery Sizing
E-004.1(Q)	Rev.13IR0	HC Class 1E 125 VDC Battery Sizing
E-009.1(Q)	Rev.06	Standby Class 1E Diesel Generator Sizing
E-010.1(Q)	Rev.01	Medium Voltage Cable Ampacity
EA-0027	Rev.01	SSW Header Pressure Loop Uncertainty
EG-0008	Rev.01	Process Setpoints for Temperature Controlling Devices for SACS
EG-0156-2	Rev.04	Safety Auxiliary Cooling System Heat Exchanger Outlet Temperature
EG-0010(Q)	Rev.03	SACS Pump Runout Trip Setpoint/NPSH Evaluation
EA-0001 (Q)	Rev.03	Station Service Water System Hydraulic Model
EA-0003	Rev.09	Station Service Water System Hydraulic Analysis
EG-0046 (Q)	Rev.02	STACS Operation
EG-0049	Rev.0	Evaluation of the Hope Creek SACS Heat Exchanger Under Increased Shell side and Tube side Flow
EG-004	Rev 01	HCGS Ultimate Heat Sink Temperature Limits

Specifications

10855-D3.9	Rev.11	Design, Installation and Test Specification for SSW
Performance Data Sheets		SACS Heat Exchangers 1A1E201, 1A2E201, 1B1E201, 1B2E201

Evaluations

H00-010	Safety Evaluation of RHR Heat Exchanger SACS Outlet Valve, 1EGV-023 Rubber Seat
H-1EG-MEE-1301, Rev 01	Engineering Evaluation of 100 F SACS Temperature Limit
H-1EG-MEE-1040, Rev 02	Engineering Evaluation of Minimum Design Temperature

Modifications

4ER-0117	10/96	Modification to Replace Lower End of Service Water Pumps
80010965	5/00	Removal of Rubber Seats for MOV 1EGHV-2512B (2EGV-026)

Drawings

E-0006-1(Q)	Rev.10	4.16 KV Class 1E Single Line Diagram
E-0009-1(Q)	Rev.16	125 Volt DC Single Line Diagram
E-0012-1(Q)	Rev.12	120 Volt Instrumentation
E-0018-1(Q)	Rev.13	480 Volt Class 1E Unit Substation Single Line Diagram
E-0025-1	Rev.10	480 Volt Unit Substation Single Line Diagram

E-0014-0	Rev.22	480 Volt Motor Control Center Tabulation
E-0021-1(Q)	Rev.15	480 Volt Motor Control Center Tabulation
E-0022-1(Q)	Rev.16	480 Volt Motor Control Center Tabulation
E-0031-1	Rev.15	480 Volt Motor Control Center Tabulation
E-0208-0(Q)	Rev.13	Station Service Water Schematic Diagram
E-0209-0(Q)	Rev.03	Station Service Water Schematic Diagram
E-0210-0	Rev.04	Station Service Water Schematic Diagram
E-0211-0(Q)	Rev.05	Station Service Water Schematic Diagram
E-0212-0(Q)	Rev.13	Station Service Water Schematic Diagram
E-0216-0(Q)	Rev.03	Station Service Water Schematic Diagram
E-0331-0(Q)	Rev.04	Station Service Water Schematic Diagram
E-0217-0(Q)	Rev.04	Safety Auxiliaries Cooling Schematic Diagram
E-0218-0(Q)	Rev.09	Safety Auxiliaries Cooling Schematic Diagram
E-0219-0(Q)	Rev.03	Safety Auxiliaries Cooling Schematic Diagram
E-0220-0(Q)	Rev.03	Safety Auxiliaries Cooling Schematic Diagram
E-0221-0(Q)	Rev.02	Safety Auxiliaries Cooling Schematic Diagram
E-0222-0	Rev.06	Safety Auxiliaries Cooling Schematic Diagram
E-0225-0(Q)	Rev.06	Safety Auxiliaries Cooling Schematic Diagram
E-0226-0(Q)	Rev.04	Safety Auxiliaries Cooling Schematic Diagram
E-0228-0(Q)	Rev.08	Safety Auxiliaries Cooling Schematic Diagram
E-0229-0(Q)	Rev.02	Safety Auxiliaries Cooling Schematic Diagram
E-0467-0(Q)	Rev.10	Safety Auxiliaries Cooling Schematic Diagram
E-0468-0(Q)	Rev.09	Safety Auxiliaries Cooling Schematic Diagram
E-0486-0(Q)	Rev.11	Safety Auxiliaries Cooling Schematic Diagram
E-4214-0	Rev.00	Traveling Screen Cable Block Diagram
J-10-0(Q)	Rev.25	Station Service Water Logic Diagrams
J-11-0(Q)	Rev.17	Safety Auxiliaries Cooling Logic Diagrams
J-105-0(Q)	Rev.09	Sequencer Fan-Out Logic Diagrams
J-4010-0(Q)	Rev.30	Station Service Water Loop Diagrams
J-4011-0(Q)	Rev.19	Safety Auxiliaries Cooling Loop Diagrams
M-10-1(Q)		Station Service Water
M-11-1(Q)		Safety Auxiliaries Cooling Water

<u>Vendor Documents</u>

VTD 316200	Controlotron Flow Meter
VTD 320901	Model 7600 Level Measurement System
VTD PE112AQ-0016	SACS Pump Motor
VTD PE112BQ-002	Service Water Pump Motor
VTD PM018Q-0366	EDG Engine Control Panel
VTD PM020Q-0031	Traveling Screen Logic
VTD PM020Q-0032	Traveling Screen Schematic
VTD PM020Q-0033	Traveling Screen Schematic

Procedures

HC.OP-AR.ZZ-0001(Q)	Attachment D1. Digital Alarm Point
HC.OP-ST.EA-0002(Q)	Service Water System Functional Test
HC.MD-GP.ZZ-0012(Q)	Electrical Component Meggering Procedure
HC.IC-DC.ZZ-0073(Q)	Bailey Millivolt Converter
HC.OP-IS-EA-001(Q)	Station Service Water Pumps - In-service Test
HCMDPMEA-0002(Q)	Service Water Intake Silt Survey and Silt Removal

Work Orders

CR990410140		1EAHV-2198C, HBC Unit Orientation
WO990402090		1EAHV-2198C/v474 C SSW Pump Discharge Valve
WO30001963		SSW Pump Motor EA-1A-P502, 18 Month PM
WO980819009		SSW Pump Motor EA-1B-P502, 18 Month PM
WO981201049		SSW Pump Motor EA-1C-P502, 18 Month PM
WO981116016		SSW Pump Motor EA-1D-P502, 18 Month PM
70007904	6/00	Traveling Screen Shear Pin Failure
60009702	6/00	“B” SSW Screen Not Running
60010759	8/00	“B” Traveling Screen Pin Sheared
70009068	8/00	“B” SSW Traveling Screen Pin Shear
971109086	2/00	“D” Screen Broken Shear Pin on 11/97
70002657	1/00	Increase in Vibration on “A” and “D” Pumps
980223094	4/98	Traveling Screen Head shaft Sprocket Replacement
971027213	4/98	Traveling Screen Pin Wear
980222086	2/98	Traveling Screen Sprocket Teeth are Loose
980313068	6/98	Broken Strands in Traveling Screen Baskets
980603122	6/98	Head Shaft Spring Needs to be Replaced

Notification Reports

20032002	6/00	“B” Service Water Screen Not Running
20036604	8/00	“B”SSW Pin Sheared
20036641	8/00	“B” SSW Traveling Screen Pin Sheared
20025355	4/00	“A” SSW Drive Sprocket Insert Broken
20008008	10/99	Tracking of PM on SW Traveling Screen
20001836`	7/99	Broken Bolt on “A” Traveling Screen
20001788	7/99	Saddle Hanging Loose on “A” Traveling Screen
20012313	12/99	“D” SSW Pump Vibration Alert
20037546	8/00	Trend Wrong DP Used in Vendor Calculation
20038165	8/00	“A” SSW Pump Failed Surveillance

System Health Reports

Station Service Water	4/1/00 - 6/30/00
Safety Auxiliaries Cooling System	10/1/99-12/31/99
Safety Auxiliaries Cooling System	4/1/00 - 6/30/00

Other Reports

Service Water Systems Operational Performance Inspection - Final Report of July 18, 1997

ATTACHMENT 2
NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues with low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, as described in the matrix. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix. More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.