July 30, 2004

EA-04-139

Mr. Mark E. Warner
Site Vice President
c/o Mr. James M. Peschel
FPL Energy Seabrook, LLC
Seabrook Station
P.O. Box 300
Seabrook, NH 03874

SUBJECT: SEABROOK STATION - NRC INTEGRATED INSPECTION REPORT 05000443/2004003

Dear Mr. Warner:

On June 30, 2004, the NRC completed an inspection at the Seabrook Nuclear Power Station. The enclosed report documents the inspection findings which were discussed on July 22, 2004, with yourself and other members of your staff.

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

This report documents an apparent violation which is being considered for escalated enforcement action in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600. The apparent violation involved a failure to properly implement 10 CFR 50.59 for a change to the Seabrook Updated Final Safety Analysis Report (FSAR). Since the change involved more than a minimal increase in the frequency of occurrence and the consequences of an accident previously evaluated in the Seabrook FSAR, the change required prior NRC approval, but approval was not requested. The NRC has not made a final determination in this matter, therefore, no Notice of Violation is being issued for this apparent violation, at this time.

Before the NRC makes its enforcement decision, we are providing you an opportunity to either: (1) respond to the apparent violation addressed in this inspection report within 30 days of the date of this letter or (2) request a predecisional enforcement conference. If a conference is held, it will be open for public observation. The NRC will also issue a press release to announce the conference. Please contact Dr. Ronald Bellamy at (610) 337-5200 within 7 days of the date of this letter to notify the NRC of your intended response.

If you choose to provide a written response, it should be clearly marked as a "Response to An Apparent Violation in Inspection Report No. 05000443/2004003; EA-04-139" and should include for the apparent violation: (1) the reason for the apparent violation, or, if contested, the basis for disputing the apparent violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate response is not received within the time specified or an extension of time has not been granted by the NRC, the NRC will proceed with its enforcement decision or schedule a predecisional enforcement conference.

In addition, please be advised that the number and characterization of apparent violations described in the enclosed inspection report may change as a result of further NRC review. You will be advised by separate correspondence of the results of our deliberations on this matter.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm.html Also, our current Enforcement Policy is included on the NRC's Web site at www.nrc.gov; select **What We Do**, **Enforcement**, then **Enforcement Policy**.

Sincerely,

/RA/

A. Randolph Blough, Director
Division of Reactor Projects

Docket No. 50-443
License No: NPF-86

Enclosure: Inspection Report No. 05000443/2004003
w/ Attachment: Supplemental Information

cc w/encl:
J. A. Stall, FPL Senior Vice President, Nuclear & CNO
J. M. Peschel, Manager - Licensing
G. F. St. Pierre, Station Director - Seabrook Station
R. S. Kundalkar, FPL Vice President - Nuclear Engineering
D. G. Roy, Nuclear Training Manager - Seabrook Station
Office of the Attorney General
P. Brann, Assistant Attorney General
M. S. Ross, Attorney, Florida Power & Light Company
R. Walker, Director, Dept. of Public Health, Commonwealth of Massachusetts
B. Cheney, Director, Bureau of Emergency Management
C. McCombs, Acting Director, MEMA
Administrator, Bureau of Radiological Health, State of New Hampshire
W. Meinert, Nuclear Engineer, Massachusetts Municipal Wholesale Electric company
J. Devine, Polestar Applied Technology
R. Backus, Esquire, Backus, Meyer and Solomon, New Hampshire
Town of Exeter
Board of Selectmen
S. Comley, Executive Director, We the People of the United States
R. Shadis, New England Coalition Staff
M. Metcalf, Seacoast Anti-Pollution League

<u>Distribution w/encl:</u> **(VIA E-MAIL)**
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H. Miller, RA
J. Wiggins, DRA
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K. Jenison, DRP
G. Dentel, SRI - Seabrook
W. Lanning, DRS
R. Crlenjak, DRS
J. Rogge, DRS
L. Doerflein, DRS
R. Conte, DRS
R. Lorson, DRS
J. White, DRS
C. Miller, EDO Coordinator
J. Clifford, NRR
S. Wall, PM, NRR
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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	05000443
License No.:	NPF-86
Report No.:	05000443/2004003
Licensee:	Florida Power & Light Energy Seabrook, LLC (FPL)
Facility:	Seabrook Station, Unit 1
Location:	Post Office Box 300 Seabrook, New Hampshire 03874
Dates:	April 1, 2004 to June 30, 2004
Inspectors:	Glenn Dentel, Senior Resident Inspector Steve Shaffer, Resident Inspector George Malone, Salem Resident Inspector Thomas Moslak, Health Physicist Jamie Benjamin, Reactor Inspector Kenneth Jenison, Senior Project Engineer Shani Lewis, Project Engineer
Approved by:	Dr. Ronald Bellamy, Chief Projects Branch 6 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000443/2004003; 04/01/2004-06/30/2004; Seabrook Station, Unit 1; Flood Protection Measures.

The report covers a 13-week period of inspection by resident inspectors and an announced inspection by a regional senior health physics inspector. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

TBD. The inspectors identified an apparent violation of 10 CFR 50.59 for implementing a change in the facility that resulted in a more than minimal increase in the frequency of occurrence and consequence of an accident previously evaluated, without obtaining NRC approval pursuant to 10 CFR 50.90. In 1997, Seabrook identified that turbine building flood diversion devices (scuppers) had not been installed in the plant as described in Seabrook's final safety analysis report (FSAR). Between 1997 and 2000, Seabrook implemented a design change which removed the turbine building scuppers from the FSAR without prior NRC approval as required by 10 CFR 50.59 and 50.90.

The inspectors determined that traditional enforcement applied because this issue impacted the NRC's ability to perform its regulatory function. The turbine building scuppers were designed to mitigate the consequences of a circulating water system failure. The system failure could create a turbine building flood, which if not addressed, could impact onsite and offsite power sources. The design change resulted in a more than minimal increase in the frequency of occurrence and consequences of a loss of offsite power accident. (Section 1R06)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

The plant began the period at full rated thermal power and operated at or near full power for the entire report period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

Full System Walkdown - Chemical Volume and Control System (CVCS) (71111.04S - 1 Sample)

The inspectors conducted a detailed review of the alignment and conditions of the CVCS. The inspectors performed a walkdown to verify the system alignment was maintained in accordance with system drawings and procedures. Control room indications were verified to be appropriate and consistent with technical specification requirements and the Updated Final Safety Analysis Report (UFSAR). The inspectors reviewed and evaluated the potential impact on system operation from open work orders, condition reports and tagged equipment. The system health report was reviewed, verified during the walkdown and discussed with the system engineer.

The inspectors reviewed the following documents to support the walkdown and to verify proper system alignment:

- Piping and instrumentation drawings (P&IDs) for the CVCS;
- A sample of historical condition reports (CRs) relative to the CVCS and its support systems (CRs 04-01665, 03-10736, and 03-08317);
- MA 4.8, "Control of Scaffolding," Rev. 7;
- MS0599.47, "Erection of Scaffolding," Rev. 0.

Partial System Walkdowns. (71111.04Q - 4 Samples)
The inspectors performed the following partial system walkdowns:

- On April 16, the inspectors performed a walkdown of the service water system while maintenance was being performed on a service water valve (SW-V-4) which provides isolation to the non-safety loads (Work Order 0216668).

- On May 12, the inspectors performed a walkdown of the service air system after the failure of the Centec service air compressor.

- On May 10 through 14, the inspectors performed a walkdown of the "B" emergency diesel generator (EDG) while the "A" EDG was out of service for scheduled maintenance.

- On May 10 through 14, the inspectors performed a walkdown of the "A" and "B" residual heat removal systems.

The inspectors conducted a walkdown of each system to verify that the critical portions of selected systems, such as valve positions, switches, and breakers, were correctly aligned in accordance with Seabrook’s procedures and to identify any discrepancies that may have had an effect on operability.

The inspectors reviewed applicable piping and instrumentation drawings and applicable operational lineup procedures to support the walkdowns and to verify proper system alignment.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope (71111.05Q - 9 Samples)

The inspectors examined several areas of the plant to assess: 1) the control of transient combustibles and ignition sources; 2) the operational status and material condition of the fire detection, fire suppression, and manual fire fighting equipment; 3) the material condition of the passive fire protection features (fire doors, fire dampers and fire penetration seals); and 4) the compensatory measures for out-of-service or degraded fire protection equipment. The following areas were inspected:

- “B” Charging Pump Room, Elevation 7'-0"
- Train ‘A’ Residual Heat Removal (RHR), Containment Building Spray (CBS), Safety Injection (SI) Equipment Vault, elevation (-) 61'
- Train “B” RHR, CBS, SI Equipment Vault, elevation (-) 61'
- Train “A” RHR, CBS, SI Equipment Vault, elevation (-) 50'
- Train “B” RHR, CBS, SI Equipment Vault, elevation (-) 50'
- Diesel Generator Building Train “A” Generator Room, elevation 21'6" & 51'6"
- Diesel Generator Building Oil Tank Rooms Train “A”, elevation (-)16'
- Turbine Floor, North & South Ends, elevation 75'
- Turbine Building Ground Floor, North & South Ends, elevation 21'

The inspectors reviewed the following documents:

- Fire Protection Pre-Fire Strategies and Fire Hazard Analysis;

- Compensatory List of Fire Protection Equipment out-of-service;
- Fire Protection Equipment Layout Drawings.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 - 3 Samples)

a. Inspection Scope

The inspectors reviewed three samples of flood protection measures. These reviews were conducted to evaluate the licensee's protection of safety-related systems from internal and external flooding conditions. The inspectors performed a walkdown of two specific internal areas of interest where licensee documents indicated increased risk significance from internal flooding events. In addition, an external walkdown was conducted. The two internal areas and the external area consisted of:

- “A” and “B” RHR pump rooms
- “A” and “B” vital switchgear rooms and turbine building condenser bays.
- Turbine building and other building exteriors

The inspectors determined whether internal and external flooding conditions were adequately addressed by Seabrook. The inspectors reviewed the Seabrook Final Safety Analysis Report (FSAR) and other design basis documents, including several flooding calculations. The inspectors compared the as-found equipment and conditions to ensure they remained consistent with those indicated in the design basis documentation, flooding mitigation documents, and risk analysis assumptions. Documents reviewed during the inspection are listed in the Attachment.

b. Findings

Introduction

The inspectors identified an apparent violation of 10 CFR 50.59 for making a change to the facility that resulted in a more than minimal increase in the frequency of occurrence and consequence of an accident previously evaluated in the Final Safety Analysis Report (FSAR) without prior NRC approval. In 1997, Seabrook determined that turbine building flood diversion devices (scuppers) had not been installed in the plant as described in the FSAR. Subsequent to this determination, Seabrook failed to properly evaluate the change to the facility in accordance with 10 CFR 50.59.

Description

When the plant was licensed in 1983, the Seabrook Safety Evaluation Report (SER), NUREG 0896, stated that Seabrook had provided an analysis of the effect of flooding on safety related equipment as a result of a postulated failure in the Circulating Water system (CW). The SER stated that the CW system had the potential for flooding the turbine building if a CW line ruptured and the CW pumps were not stopped. The SER further stated that continued operation of the pumps would cause water to flow out of the turbine building through scuppers and doors to the yard and away from plant buildings. Shutdown of the pumps would eventually stop the flow. The SER concludes that a total failure in the CW system "would not result in flooding which would compromise plant safety." The Seabrook SER further stated that water level alarms were installed in the CW pits in the turbine building that would alert the control room operators in the event of a CW system rupture. The NRC staff concluded that the "circulating water system [met] the requirement of GDC 4 with respect to protection of safety related systems," and that it met the acceptance criteria of SRP 10.4.5. The FSAR mirrored the wording in the SER. In 1997, the FSAR stated that the "scuppers ... [in] the turbine building will allow the water to flow out of the building, preventing excessive water build up on the building floor. ... the [CW] pit would fill up ... unless prompt action by the operator is taken. No safety-related equipment is affected by a failure of this equipment."

In 1997, Seabrook identified that components in the turbine building and the CW system that were part of the original design basis, as described in the FSAR, Section 10.4.5, were never installed. These components include turbine building flood diversion devices (scuppers) and CW pit level switches. In 1997, the inspectors evaluated the non-conformance to the FSAR and issued a non-cited violation (see NRC Inspection Report 05000433/97-06). In response to this issue, Seabrook completed a design change which installed the level switches and removed the description of the scuppers in the FSAR through a 10 CFR 50.59 evaluation. Seabrook determined that the change did not need NRC approval.

The inspectors reviewed the 1997 design change (DRC 97-0033) and its associated 10 CFR 50.59 evaluation and determined that the change to eliminate the scuppers resulted in a more than minimal increase in the frequency of occurrence and the consequences of an accident previously evaluated in the FSAR and thus needed prior NRC approval. Therefore, the change was a violation of 10 CFR 50.59. The inspectors performed this evaluation using both the 10 CFR 50.59 regulation in existence in 1997 and the subsequent revision to the regulation in 1999. NEI Guidance 96-07, Guidelines for 10 CFR 50.59 Implementation, November 2000, Revision 1, defines and provides examples of what conditions constitute more than minimal increases of risk. Based on the guidance, the inspectors concluded that the removal of the turbine building scuppers used for mitigating a turbine building flood constituted a more than minimal increase in risk. The flooding of the turbine building potentially impacts the turbine building relay room (offsite power sources) and the emergency switchgear rooms (onsite power sources). The inspectors also identified that Seabrook had two opportunities to identify the violation during closeout review of design change notices (DCN) and during their reexamination of the 10 CFR 50.59 evaluation associated with the design change.

Analysis.

- Screening for "Old Design Issues"

  NRC Inspection Manual Chapter (IMC) 0305 allows credit to be given to the licensee for self identification of certain "old design issues," such as engineering calculations, engineering analyses, associated operating procedures or plant equipment installations, if all four of the criteria in IMC 0305 are met. The inspectors determined that two of the four "old design criteria" were not met in the case of the 1997 facility change: (1) The licensee's failure to meet 10 CFR 50.59 requirements in 1997 and in 2000 were identified by the NRC and the NRC review of the facility change has not yet been performed; (2) The inspectors determined that there were performance deficiencies associated with the 1997, 10 CFR 50.59 and with the 2000 design change notice (DCN 1-3) in that the 2000 DCN affirmed, incorrectly, that the 1997 review was adequate.

- Screening for Traditional Enforcement

  In accordance with Inspection Manual Chapter (IMC) 0612, Appendix B, "Issue Disposition Screening," the inspectors determined that traditional enforcement applied because this issue potentially has impacted the NRC's ability to perform its regulatory function. Specifically, the licensee failed to perform an adequate 10 CFR 50.59 analyses and failed to obtain a license amendment pursuant to 10 CFR 50.90, in 1997 as required by 10 CFR 50.59(c), when it was discovered that the turbine building, as-built, had been changed from the facility as described in the final safety analysis report (FSAR) [as updated]. The licensee also departed from the method of evaluation described in FSAR Section 10.4.5 regarding the impact of non-safety-related systems on safety-related systems used to establish that the design basis of the circulating water (CW) system met GDC 4. NRC IMC 0612 provides for a risk assessment to this traditional enforcement process when NRC considers it appropriate. The change increased the likelihood of loss of offsite power and loss of vital buses initiating events. The elimination of the turbine building internal flood mitigating scuppers resulted in a more than minimal increase in the frequency of occurrence [10 CFR 50.59(c)(2)(i)] of the loss of offsite power accident previously evaluated in FSAR Section 15.2.6.

- Phase III of the NRC Significance Determination Process

The Regional Risk analyst performed a Phase 3 SDP using the Seabrook, Rev 3, Standardized Plant Analysis Risk (SPAR) Model, Level 1, Change 3.01, created January 2004, that was updated using NUREG-5496 data regarding loss of offsite power (LOOP) initiating event (IE) frequency, recovery probabilities and emergency diesel generator (EDG) mission times. Common cause failure alpha factors were updated using information that the Idaho National Engineering and Environmental Laboratory (INEEL) staff and the Nuclear Regulatory Commission's (NRC) Office of Nuclear Regulatory Research have developed from data collection and analysis of common cause failure events from 1980 through 2001.

The analysis assumed that if the plant had been constructed in accordance with the design and licensing basis as described in FSAR section 10.4.5 (before update in 1997), there would be a minimal contribution to plant risk, core damage probability or loss of offsite power frequency due to a CW system rupture and a turbine building flood.

Without the internal flood diversion scuppers installed, the failure of a CW expansion joint and flooding of the turbine building could result in three potential conditions: (1) a non-recoverable loss of offsite power (LOOP); (2) a non-recoverable LOOP combined with the loss of one vital 4kV AC bus; (3) a non-recoverable LOOP combined with the loss of both vital 4kV AC buses.

The analyst calculated the initiating event frequency for each of these three flooding related events using the method described in Section 12.1, Internal Flooding, of the 2002 Seabrook Probabalistic Safety Study (SSPSS-2002). The analyst accepted some of the licensee’s inputs for this calculation method including the large and very large internal flooding initiating event frequencies, the probability values for the flood to propagate into each of the two vital switchgear rooms, and the probability that the switchgear rooms’ (75 gpm max) floor drains would mitigate and prevent the flooding from affecting the two vital 4kV AC trains.

The analyst independently calculated the human error probability inputs using the NRC SPAR Model Human Error (HE) Worksheets. SSPSS-2002 documented that the times to the loss of offsite power due to flooding of the offsite power relay room in the turbine building were 8 and 40 minutes for very large and large floods, respectively. However, in response to this issue, the licensee recalculated the flooding rate and determined that the times to the loss of offsite power due to large and very large floods are 18 and 92 minutes. The staff used these more recently-calculated times to complete the SPAR HE Worksheets. Use of these times resulted in the performance shaping factors (PSFs)for the available diagnosis times to be considered “barely adequate” and “extra” for the very large and large floods, respectively. Although the inspectors raised issues regarding the adequacy of the flooding alarms, adequacy of the alarm response and emergency operating procedures, and the stress and complexity associated with the required actions, the staff analysis set all other PSFs to nominal values. The human error probabilities (HEP) were dominated by the very large flood and were calculated to be in the low E-1 range using the SPAR HE Worksheets. Alternatively, if the SPSS-2002 time (8 minutes) to loss of offsite power for a very large flood was used in the HRA calculation, the HEP value would default to 1.0 (one order of magnitude higher) due to inadequate time diagnoses and response to the event. Using the SPAR risk assessment tools and the 18 and 92 minute assumptions, the frequency per year (and the increase in frequency above the initial minimal value) of the non-recoverable loss of offsite power due to a large or very large flood was calculated in the low E-4 range. Therefore, the staff determined that the change to the facility from that described in the pre-1997 FSAR resulted in a more than minimal increase in the frequency of occurrence of the loss of offsite power accident previously evaluated in the FSAR.

The licensee used the HCR\ORE\THERP method from the EPRI HRA calculator and calculated an HEP value, in the low to mid E-2 range, for the operator action to open the turbine building roll-up door to provide a flood drainage path and mitigate the very large flood event. This calculation assumed “time pressure” and “skill-based” response performance shaping factors. The licensee’s HRA analysis also assumed simple response with very good cues and indication, operations at control room panels, and low stress. Considering the possibility to recover from the inability to open the turbine building door by stopping the circulating water pump as the recovery method, the licensee’s calculated HEP value would drop (an order of magnitude) to the low to mid E-3 range.

For the first event (a non-recoverable loss of offsite power due to a large or very large flood), an initiating event assessment was performed by setting the LOOP initiating event (IE) and the failure to recover offsite power probabilities to “true” (or 1.0). All other initiating event probabilities were set to 0 and a conditional core damage probability (CCDP) was calculated. The product of the CCDP and the frequency (per year) of the flood induced non-recoverable LOOP event was calculated to determine the change in core damage frequency ($\triangle$CDF). A similar assessment was performed for the two remaining events and the sum of the $\triangle$CDF values for the three turbine building flooding events was determined to be in the low E-6 range. Therefore, the change to the facility from that described in the pre-1997 FSAR represented a more than minimal increase in the consequences of the loss of offsite power accident previously evaluated in the FSAR.

Enforcement

10 CFR 50.59 (c) states in part, that the licensee may make a change in a facility as described in the final safety analysis report (FSAR) without prior Commission approval, provided the proposed change does not involve more than a minimal increase in the frequency of occurrence and the consequences of an accident previously evaluated in the FSAR. NRC Regulatory Guide 1.187, Guidance for Implementation of 10 CFR 59, provides guidance on the implementation of 10 CFR 50.59 and endorses (with exceptions) Nuclear Energy Institute (NEI) 96-07,Guidelines for 10 CFR 50.59 Implementation, November 2000, Revision 1. NEI 96-07 defines and provides examples of conditions that constitute more than minimal increases of risk.

Contrary to the above, Seabrook completed a change to the facility that represented more than a minimal increase in the frequency of occurrence and the consequences of a previously evaluated accident without prior NRC approval. Between 1997 and 2000, Seabrook implemented DCR 97-0033 and design change notices 01 through 03 which removed the turbine building scuppers from Seabrook's FSAR.

This violation of requirements is being treated as an apparent violation of 10 CFR 50.59 (c), 05000443/2004-003-001, Failure to Obtain Prior NRC Approval for a Change to the Facility.

1R07 Heat Sink Performance (71111.07A - 2 Samples)

a. Inspection Scope

The inspectors reviewed two samples of safety related heat exchangers to identify any degraded performance or potential for common cause problems that could increase plant risk. The inspectors reviewed the results of recent residual heat removal (RHR) system and component cooling water (CCW) system health reports and ensured that associated performance data were documented and met the design performance criteria. In addition, the inspectors compared the most recent performance data of the RHR and CCW heat exchangers with the trend and system data in the system health report. The inspectors also reviewed the Final Safety Analysis Report to ensure that RHR heat exchanger performance criteria were consistent with the Seabrook design basis. The inspectors verified that adverse conditions and work orders documented in the RHR system health report were appropriately entered into Seabrook's corrective action program and adequately addressed.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

Quarterly Resident Inspector Review (71111.11Q - 1 Sample)

a. Inspection Scope

On June 14, the inspectors observed an operator training session focusing on human performance of time critical tasks. The inspectors reviewed the operators' abilities to correctly evaluate the training scenario and implement the emergency plan. Operator actions were reviewed against Seabrook's procedural requirements. The inspectors also evaluated whether deficiencies were identified and discussed during critiques.

b. Findings

No findings of significance were identified

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope (71111.12Q - 2 Samples)

The inspectors completed two maintenance rule samples including one system review and one specific issue review.

System Review

The inspectors evaluated Maintenance Rule (MR) implementation for the diesel air handling system. The system was categorized in 10 CFR 50.65(a)(1) due to repetitive failures of the starters for the diesel air handling fans on the 480 VAC motor control centers. The inspectors interviewed engineers, reviewed specific maintenance rule criteria for the 480 VAC motor control centers, and examined the apparent cause determination and corrective actions of CR 03-07222. The inspectors reviewed the (a)(1) improvement plan and system monitoring plan and evaluated the activities against 10 CFR 50.65.

Maintenance Rule Functional Failure (MRFF) Review

The inspectors reviewed the application of the maintenance rule for a temporary loss of seal injection flow during the performance of procedure OS1003.03. The inspectors conducted interviews, reviewed the Updated Final Safety Analysis Report (UFSAR), specific maintenance rule criteria and the system health report for the CVCS system. Additionally, the inspectors reviewed the associated apparent cause for condition report (CR 03-08317) and assigned corrective actions. The inspectors compared the maintenance rule functional failure evaluation against 10 CFR 50.65 requirements and against the guidance in NUMARC 93-01, "Industry Guideline for Monitoring the effectiveness of Maintenance at Nuclear Power Plants," Rev. 2. Based on the inspection, CR 04-04903 was generated to reevaluate the maintenance rule functional failure determination.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13 - 5 Samples)

a. Inspection Scope

The inspectors reviewed the scheduling and control for two planned maintenance activities and three emergent work troubleshooting activities in order to evaluate the effect on plant risk. The inspectors conducted interviews with operators, risk analysts, maintenance technicians, and engineers to assess their knowledge of the risk associated with the work, and to ensure that other equipment was properly protected. The inspectors evaluated the compensatory measures against Seabrook procedures, Maintenance Manual 4.14, "Troubleshooting," and Work Management Manual 10.1, "On-Line Maintenance." Specific risk assessments were conducted using Seabrook's "Safety Monitor." The inspectors reviewed the following items:

- On May 19 and 20, the inspectors reviewed the plant risk configuration for maintenance on the "A" emergency feedwater pump and one switchyard breaker;

- On April 7 and 8, the inspectors reviewed the on-line maintenance assessment for troubleshooting work on the slow flow solenoid for MS-V-88. The inspectors observed portions of the work activity, examined the work order (WO) 0415025 and associated documents, and interviewed the maintenance technicians. The work documents were evaluated against various Seabrook procedures including Work Management Manual (WM) 8.4, "Work Control Practices," Rev. 2;

- On May 19 and 20, the inspectors reviewed the on-line maintenance assessment for troubleshooting work for the steam driven emergency feedwater pump due to higher than expected temperatures on the outboard bearing of the pump. The inspectors observed portions of the work activity, reviewed WOs 0340705, 0419680, and 0419691. The inspectors also interviewed engineers, maintenance technicians and operators;

- On May 10 to 14, the inspectors reviewed the plant risk configuration during the "A" EDG maintenance outage. The inspectors also evaluated the emergent activities associated with high jacket water temperature instrument failure and an inadvertent auxiliary fuel oil pump auto start.

- On June 22 to 24, the inspectors reviewed the on-line maintenance assessment for troubleshooting work for the Containment Enclosure Ventilation Area (CEVA) to atmosphere differential pressure instrumentation. The instruments required rescaling to be able to measure the required differential pressure due to a change in calculation methodology. The inspectors observed portions of the work activities and reviewed WOs 0423283, 0423284, 0423344, and Design Change Request (DCR) 04DCR008. The inspectors interviewed engineers, I&C technicians, and operators involved in the operation.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance Related to Non-Routine Plant Evolutions and Events (71111.14 - 2 Samples)

a. Inspection Scope

The inspectors reviewed operator response to two non-routine evolutions.

Main Feedwater Pump Issue
The inspectors reviewed operator performance in response to increasing vibration levels on the “B” main feedwater pump FW-P-32-B. The inspectors verified that operators evaluated the increasing vibration and took appropriate actions to address the condition in accordance with procedures. The “A” main feedwater pump was biased under WO 0412384 using procedure ON1035.10, “Main Feed Pump Standby and Start Up Operation,” Rev. 7.

Circulating Pump Trip
On May 18, the "C" circulating water pump tripped due to a human performance error during a maintenance activity. The unit remained at full power as the two remaining circulating water pumps maintained sufficient flow to the plant. The inspectors reviewed operator performance in response to the loss and subsequent restart of the pump. The inspectors examined operator response against alarm response procedures, "CW Pump C Breaker Trip and L/O," and ON 1238.01, "Circulating Water Screens Fouled Abnormal," Rev. 5. The inspectors reviewed operator actions to restart the pump against operating procedures, ON 1038.01, "Circulating Water System Pump Startup," Rev. 7 and ON 1017.02, "Circulating Water Screen Wash Operation," Rev. 5.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 4 Samples)

a. Inspection Scope

The inspectors reviewed operability evaluations and/or condition reports in order to verify that the identified conditions did not adversely affect safety system operability or plant safety. The evaluations were reviewed using criteria specified in Generic Letter 91-18, "Resolution of Degraded and Nonconforming Conditions" and Inspection Manual Part 9900, "Operable/Operability - Ensuring the Function Capability of a System or Component." In addition, where a component was determined to be inoperable, the inspectors verified that Technical Specifications (TS) limiting condition for operation implications were properly addressed. The inspectors performed field walkdowns, interviewed personnel, and reviewed the following items:

- CR 04-03519, which evaluated degraded or non-conforming conditions on the safety related 4160 volt breakers to ensure that operability was justified and that mitigating systems or those affecting barrier integrity remained available. The inspectors reviewed licensee performance to ensure all related TS and FSAR requirements were met.

- CR 04-04438, which evaluated the impact of a failed containment temperature input which was removed from alarm but remained in the computer calculation for average containment temperature. This calculation is used to satisfy the TS surveillance requirement 4.6.1.5. The inspectors reviewed the apparent cause and corrective actions, interviewed operators and system engineers, examined procedure ON 1090.06, "Use and Control of Deleted Analog and Digital Points," Rev. 3, and reviewed associated CRs (03-04884 and 04-05778) and WOs (0319738, 0319743, and 0422593). The inspectors also reviewed past containment temperature data to determine whether the TS maximum average temperature had been exceeded.

- CRs 04-04086, 04-04002 and 04-04010, which evaluated the impact of the residual heat removal (RHR) breaker not capable of being racked out to the test position. Operators attempted to rack out the breaker but encountered interference with a metal raceway in the breaker cubicle. The inspectors reviewed the operability evaluation, conducted independent walkdowns of the RHR and other 4kV breakers, and interviewed several engineers.

- CR 04-04780, which evaluated the impact of increasing emergency feedwater pump outboard bearing temperature. The inspectors observed portions of multiple surveillance tests, reviewed the operability evaluation, and examined the detailed temperature trend data.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16 - 1 Sample)

a. Inspection Scope

The inspectors completed a review of one specific operator workaround.

The inspectors reviewed the staging of flashlights at the control panels as a compensatory measure due to Inverter ED-I-9 being inoperable. In a station blackout with ED-I-9 being inoperable, the control room would lose overhead lighting. The inspectors reviewed CR 04-05505 and Standing Operating Order 04-016 in order to evaluate the impact on operators.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19 - 5 Samples)

a. Inspection Scope

The inspectors reviewed post-maintenance testing (PMT) activities to ensure: 1) the PMT was appropriate for the scope of the maintenance work completed; 2) the acceptance criteria were clear and demonstrated operability of the component; and 3) the PMT was performed in accordance with procedures. The following PMTs were reviewed:

- On April 27, the activities associated with the repair of the "B" Charging Pump discharge vent piping. The inspectors observed portions of the maintenance activity, interviewed maintenance technicians and operators, and reviewed WO 0336809.

- On May 21 and 24, the retests described in WO 0418722 were performed following replacement of the “A” Electrohydraulic Control Pump. The inspectors reviewed the test results and the work order.

- On June 2, the torque checks performed on the manifold bolts on feedwater transmitters (1-FW-FT-543, 1-FW-FT-513, 1-FW-FT-533, 1-FW-FT-512, and 1-FW-FT-532). The inspectors observed the torque checks and reviewed WOs 0420265 through 0420269.

- On June 16 and 17, the torque checks performed on the manifold bolts on nine feedwater transmitters (1-FW-LT-501, 1-FW-LT-4310, 1-FW-LT-502, 1-FW-LT-4320, 1-FW-FT-523, 1-FW-LT-503, 1-FW-LT-4330, 1-FW-LT-504, and 1-FW-LT-4340). The inspectors observed the torque checks and reviewed WOs 0421228 through 0421236.

- On June 11, fuse 19, associated with a shutdown control rod, and the blown fuse indicator for this fuse were replaced. The inspectors observed the fuse replacement, interviewed maintenance technicians, and observed the initial thermography of the replacement fuse and the blown fuse indicator. The inspectors also reviewed WO 0421227.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 5 Samples)

a. Inspection Scope

The inspectors observed portions of surveillance testing activities of safety-related systems to verify that the system and components were capable of performing their intended safety function, to verify operational readiness, and to ensure compliance with required Technical Specifications and surveillance procedures.

The inspectors attended some of the pre-evolution briefings, performed system and control room walkdowns, observed operators and technicians perform test evolutions, reviewed system parameters, and interviewed the system engineers and field operators. The test data recorded was compared to procedural and technical specification requirements, and to prior tests to identify any adverse trends. The following surveillance procedures were reviewed.

- On May 19, OX1436.02, “Turbine Driven Emergency Feedwater Pump Quarterly and Monthly Valve Test,” Rev. 8;
- On June 10, OX1416.05, "Service Water Cooling Tower Pumps Quarterly and 2 Year Comprehensive Test," Rev. 7. The inspectors conducted an in-office review of the completed surveillance test;
- On June 16, OX1423.07, “Containment Enclosure Emergency Exhaust Filter System 31 Day,” Rev. 6;
- On June 18, OX1426.17, “DG 1B Tech Spec Action Statement Surveillance,” Rev. 4; and
- On June 24, OX1430.04, "Main Steam System Valve Operability Tests," Rev. 3. The inspectors conducted an in-office review of the completed surveillance test.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23 - 1 Sample)

a. Inspection Scope

The inspectors reviewed a plant modification to determine whether it met the criteria of a temporary modification or temporary alteration. The modification involved the installation of service air tubing with isolation and drain valves to provide service air system pressure at the inlet of the portable air compressor to facilitate auto-start capabilities of the temporary air compressors.

The inspectors interviewed engineers and operators, completed field walkdowns, and reviewed the Temporary Modification Request, 2004-004, Rev. 01 and WO 0418708.

The inspectors verified that the equipment was installed in accordance with NRC requirements and plant procedures. The inspectors also examined the combined effect of the modification with the outstanding temporary modifications.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06 - 1 Sample)

a. Inspection Scope

The inspectors reviewed emergency classification and notification completed by operators during requalification training on June 14 (See Section 1R11). The inspectors evaluated the results against Seabrook's Emergency Response Manual 1.1, "Classification of Emergencies" and NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Rev. 2.

b. Findings

No findings of significance were identified.

1. RADIATION SAFETY

Occupational Radiation Safety [OS]

2OS1 Access Control to Radiologically Significant Areas (71121.01 - 21 Samples)

a. Inspection Scope

On May 17 to 20, the inspectors verified that Seabrook was properly implementing physical, administrative and engineering controls for access to locked high radiation areas, and other radiologically controlled areas during power operations, and that workers were adhering to these controls when working in these areas. Implementation of these controls was reviewed against the criteria contained in 10 CFR 20, applicable industry standards, and Seabrook's procedures.

Plant Walkdown and RWP Reviews

The inspectors identified exposure significant work areas including areas in the Waste Handling Building, Containment Building, Primary Auxiliary Building, and Fuel Handling Building. Tasks in the Waste Handling Building included transfer of a spent resin liner from the storage area into a shipping cask and preparation of the cask for shipment. Tasks in the Containment Building included accumulator sampling, boric acid cleaning, and ECCS valve verification. Tasks in the Primary Auxiliary Building included removal and transfer to storage of a spent fuel pool filter (SFP-F-33). Tasks in the Fuel Handling Building included inspection of wall surfaces in the fuel transfer canal and testing of fuel transfer equipment. The inspectors reviewed the radiation work permits (RWP) and the radiation survey maps associated with these work areas to determine whether the radiological controls were acceptable.

The inspectors toured accessible radiological controlled areas, and with the assistance of a radiation protection technician, performed independent radiation surveys of selected areas to confirm the accuracy of survey data and adequacy of postings.

In reviewing RWPs, the inspectors reviewed electronic dosimeter dose/dose rate alarm set points to determine if the set points were consistent with the survey indications and plant policy. The inspectors verified that the workers were knowledgeable of the actions to be taken when the electronic dosimeter alarms or malfunctions for tasks being conducted under selected RWPs. Work activities reviewed included spent resin liner handling (RWP 04-R-00020, Task 1), various tasks performed in the Containment Building during power operations (RWP 04-R-00010, Tasks 1, 2, 3, 4), removal/transfer of a spent fuel pool filter (RWP 04-R-00013, Task 2), and fuel transfer canal inspection (RWP 04-R-00026, Tasks 1, 2).

The inspectors reviewed various RWP and associated instrumentation, respiratory protection, and engineering controls for potential airborne radioactivity areas. Through review of relevant documentation and discussions with cognizant plant staff, the inspectors confirmed that no worker received an internal dose in excess of 50 mrem due to airborne radioactivity since the last inspection.

The inspectors reviewed the physical and administrative controls for highly contaminated materials stored in the spent fuel pool.

Problem Identification and Resolution

The inspectors reviewed elements of Seabrook's Corrective Action Program related to controlling access to radiologically controlled areas, to determine if problems were being entered into the program for resolution. Details of this review are contained in Section 4OA2 of this report.

Jobs-In-Progress

The inspectors observed aspects of various maintenance and operational activities being performed during the inspection period to verify that radiological controls, such as required surveys, area postings, job coverage, and pre-job RWP briefings were conducted; personnel dosimetry was properly worn; and that workers were knowledgeable of work area radiological conditions. Tasks observed were selected aspects of transferring a spent resin liner to a shipping cask, a containment entry for accumulator sampling, removal/transfer of a spent fuel pool filter, and fuel transfer canal inspections.

High Risk Significant, High Dose Rate HRA and VHRA Controls

The inspectors discussed with the Health Physics Supervisor the controls and procedures pertaining to High Dose Rate (HDR) areas and Very High Radiation Areas (VHRA). The inspectors verified that any changes to relevant Seabrook procedures did not substantially reduce the effectiveness and level of worker protection. Controls for significant high risk areas that were reviewed included an entry into the containment building during power operations and inspections in the fuel transfer canal.

The inspectors discussed with senior radiation protection technicians the controls in place for special areas that have the potential to become VHRA during certain plant operations. These special areas include the fuel transfer canal and Containment Building during power operations. The inspectors verified the prerequisite radiation protection department's communications and controls were in place to allow completion of timely actions, such as properly posting and controlling access to affected areas.

Keys to Locked High Radiation Areas (LHRA) and Very High Radiation Areas (VHRA), maintained at the health physics control point and in the control room, were inventoried, and accessible LHRAs were verified to be properly secured and posted during plant tours.

Radiation Worker/Radiation Protection Technician Performance

The inspectors observed radiation worker and radiation protection technician performance by attending various pre-job RWP briefings and morning staff meetings.

The inspectors reviewed condition reports related to radiation worker and radiation protection errors to determine whether an observable pattern traceable to a similar cause was evident.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification (71151 - 3 Samples)

The inspectors sampled licensee submittals for the performance indicators (PIs) listed below for the period from April 2003 through March 2004. PI definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Rev. 2 were used to verify the accuracy of the PI data reported during that period and the basis in reporting for each data element.

Mitigating Systems Cornerstone

- Safety System Unavailability, High Pressure Safety Injection Systems
- Safety System Unavailability, Emergency AC Power
- Safety System Unavailability, Heat Removal System (Emergency Feedwater)

The inspectors reviewed operator logs, surveillance tests, condition reports, system health reports and other relevant documents, and interviewed applicable licensee personnel to verify the accuracy and completeness of Seabrook's PI data. The inspectors also reviewed the accuracy of the number of required/critical hours reported.

4OA2 Identification and Resolution of Problems (71121.01)

1. Access Control to Radiologically Significant Areas

a. Inspection Scope

The inspectors reviewed twelve CRs, recent Radiation Safety Committee meeting minutes, a Nuclear Oversight Audit Report (SBK-04-01), Daily Quality Summary Reports, and materials used in presenting the As Low As Reasonably Achievable (ALARA) Plan for the next refueling outage, to evaluate Seabrook's threshold for identifying, evaluating, and resolving occupational radiation safety problems. This review included a check of possible repetitive issues such as radiation worker and radiation protection technician errors.

The review was conducted against the criteria contained in 10 CFR 20, Technical Specifications, and Seabrook's procedures.

b. Findings

No findings of significance were identified.

2. Problem Identification and Resolution Trend Review (71152 - 1 sample)

a. Inspection Scope

The inspectors reviewed Seabrook's corrective action program to identify trends that may indicate existence of more safety significant issues. The inspectors reviewed the corrective action database through the review of individual components to identify equipment degradation trends. Additionally, the inspectors reviewed Seabrook's programs for identifying trends through their performance improvement group, the individual departments, and the condition report oversight group. The inspectors also reviewed several trend condition reports.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

1. TI 2515/156, Offsite Power System Operational Readiness
Cornerstones: Initiating Events, Mitigating Systems

a. Inspection Scope

The inspectors performed Temporary Instruction 2515/156, "Offsite Power System Operational Readiness." The inspectors collected and reviewed information pertaining to the offsite power system specifically relating to the areas of the maintenance rule (10 CFR 50.65), the station blackout rule (10 CFR 50.63), offsite power operability, and corrective actions. The inspectors reviewed this data against the requirements of 10 CFR 50 Appendix A General Design Criterion 17, *Electric Power Systems*, 10 CFR 50.65 (a)(4), and Plant Technical Specifications.

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

Exit Meeting Summary

The inspectors presented the inspection results to Mr. M. Warner on July 22, 2004, following the conclusion of the period. The licensee acknowledged the findings presented. The licensee did not indicate that any of the information presented at the exit meeting was proprietary.

Site Management Visit

On June 18, Mr. Hubert Miller, Regional Administrator, US NRC Region I, and Mr. Richard Crlenjak, Deputy Division Director, Division of Reactor Safety, toured the site and met with Mr. Mark Warner and other members of Seabrook's management.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

P. Allen	Senior Health Physics Technician
M. Bianco	Supervisor, Radiological Waste Services
L. Bladow	Manager, Nuclear Oversight
R. Campion	Nuclear Oversight Auditor
W. Cash	Health Physics Department Manager
D. Cormier	Senior Health Physics Technician
T. Date	Senior Health Physics Technician
P. Dundin	Shift Operations Manager
D. Flahardy	Senior Health Physicist
D. Hampton	Supervisor, Health Physics
L. Johnson III	Senior Health Physics Technician
M. Kiley	Operations Manager
P. Nardone	Reactor Engineer
M. O'Keefe	Regulatory Compliance Supervisor
J. M. Peschel	Manager - Licensing
M. Scannell	Supervisor, Health Physics
R. Sterritt	Senior Nuclear Analyst, ALARA
M. Sullivan	Senior Health Physics Technician
R. Thurlow	Health Physics Technical Supervisor
J. Watts	Nuclear Oversight Auditor

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened:

05000443/2004003-001	AV	Failure to Obtain Prior NRC Approval for a Change to the Facility

Closed:

None.

Opened and Closed

None.

Discussed

None.

LIST OF DOCUMENTS REVIEWED

Section 1R06 - Flood Protection Measures

Documents

NRC Safety Evaluation Report (SER) NUREG 0896
Individual Plant Examination for Seabrook Station, dated March 1991
FPL Seabrook Engineering Evaluation 90-05
FPL Seabrook Memo (O'Keefe/Robertson) dated 5/10/04
FPL Seabrook Report SB2002X Model SSPSS 2002 dated 2/5/03
General Rubber memo (Aanonsen/Heckscher) dated 1/18/00
General Rubber Technical data sheets and test reports, dated 18/08/00
Circulating Water System Description dated 6/14/93
ACR 97-21283, No Scuppers are Install in the Turbine Building
ACR 00-1898, dated 2/16/2001
DCR 97 -0033 - Condenser Pit Flood Level Switch, 10/15/97
DCN 01, 02 and 03 to DCR 97-0033, closed out on 10/16/00
Calculation C-X-1-21802, Expansion Joint Rupture in the Circulating Water System Located in the Turbine Building, dated October 1997
Calculation United Engineers Study of a Rupture of Circulating Water Expansion Joints. dated February 1974
10 CFR 50.59 Evaluation , Condenser Pit Flood Level Switch, dated October 1997
NRC Safety Evaluation Report (SER) NUREG 0896
FPL Seabrook email (O'Keefe/Robertson) dated 5/10/04
United Engineers and Constructors Specification for GE Installation for Package 21

Alarm Response Procedures and Level Instrumentation Data

Level Switch 1-DF-LSHH-5985/computer point D6688
Level Switch 1-DF-LSHH-598xx/computer point D8433
Alarm Response D6688 for Level Switch 1-DF-LSHH-5985
Alarm Response D8433 for Level Switch 1-DF-LSHH-598x

Work Orders

0219738	Condenser Pit Level
0233612	Main Condenser Water box

Condition Reports (CR)

00-01898	Contrary to UFSAR Section 10.4.5.3, There are No Scuppers in the East Wall of the Circulating Water Pump House
00-04854	Contrary to UFSAR Section 10.4.5.3, There are No Scuppers in the East Wall of the Circulating Water Pump House
00-09273	Corrective Actions for UFSAR update

Sections 1R13 and 1R15 - Operability Determination and Maintenance Activity

Documents

On-Line Maintenance Plan, Diesel Generator (DG) "A" LCO - May 10 and 12, 2004

Work Orders

0339314	Current Injection Testing of DG_P-122A
0339249	Switchgear Breakers Trip Check
0339292	Line Breaker Differential Relay - SA-1 Inoperable
0339330	Lube Oil Inlet Temperature
0339329	EDG Prelube Oil Temperature Switch
0319915	Rocker Arm Lube Pumps
0339312	EDG Coolant Backup Pump
0231219	EDG Relief Valve 1-DG-V-62-A
0336213	EDG 4160 Volt Breaker Repair
0413751	EDG 4160 Volt Breaker Inspection and Repair
91D0239	EDG 1-DG-TT-7-A2 Repair
00C5708	EDG DG-1A Engine Air Cooler CCW Temperature
0417332	EDG Fuel Oil Receipt
0413126	1-DG-1-A 4160 Volt Breaker Inspection and Repair
0336216	1-DG-1-A 4160 Volt Breaker Inspection and Repair
0339292	1-DG-1-A 4160 Volt Breaker Inspection and Repair

Condition Reports

CR 03-05177	Standby Conditions for EDG
CR 04-01942	EDG Exhaust Corrosion
CR 04-02327	EDG Engine Signature Analysis
CR 04-04504	EDG Biobor out of specification high
CR 04-03519	EDG ED-X-3B Breaker Arching Contact

Other References

Colt Pielstick PC-2V Drawing 004-020

Alarm Response Procedure D6560.pro DG "A" AUX FUEL Oil Pump Running

FSAR Section 9.5	EDG Fuel Oil Storage and Transfer System
Repetitive Activity	95RM1136500 RAT XFMR 1-ED-X-3B
Repetitive Activity	98RM44843001 XFMR 1-ED-X-3B
Repetitive Activity	99RM17414001 XFMR 1-ED-X-3B
Repetitive Activity	97RM44594001 XFMR 1-ED-X-3B
Repetitive Activity	97RM41281001 4160 Breaker control circuit inspection
Repetitive Activity	97RM11365001 4160 Breaker control circuit inspection
Repetitive Activity	98RM44842001 4160 Breaker control circuit inspection
Repetitive Activity	99RM44907001 4160 Breaker control circuit inspection
Repetitive Activity	97RM44718001 4160 Breaker control circuit inspection
Repetitive Activity	97RM44719001 4160 Breaker control circuit inspection
Repetitive Activity	97RM17605001 4160 Breaker control circuit inspection

Section 2OS1: Access Control to Radiologically Significant Areas

Procedures

HD0958.03, Rev 23	Personnel Survey and Decontamination Techniques
HD0958.17, Rev 12	Performance of Routine Radiological Surveys
HD0958.30, Rev 23	Inventory and Control of Locked or Very High Radiation Area Keys and Locksets
HD0963.02, Rev 13	Administrative Guidelines for Health Physics Instrumentation
HD0992.02, Rev 28	Issuance and Control of Personnel Monitoring Devices
HN0951.04, Rev 06	Health Physics Repetitive Tasks
HN0958.13, Rev 25	Generation and Control of Radiation Work Permits
HN0958.25, Rev 25	High Radiation Area Controls
HN0958.30, Rev 23	Inventory and Control of Locked or Very High Radiation Area Keys and Locksets
HN0958.39, Rev 04	Multi-Badge Control & Exposure Tracking
JD0999.910, Rev 0	Reporting Key Performance Indicators
RP 2.1, Rev 18	General Radiation Worker Instruction and Responsibilities
RP 9.1, Rev 17	RCA Access/Egress Requirements
RP 9.2, Rev 8	Radiological Access Requirements to Containment Area
RP 13.2, Rev 4	Storage of Highly Radioactive Material in the Reactor Cavity or Spent Fuel Pool
RP 15.1, Rev 17	Job Pre-Planning and Review for Radiation Exposure Control
RP 15.2, Rev 09	ALARA Recommendations
RP 15.4, Rev 10	Use and Control of Temporary Shielding
RP 15.5. Rev 03	Exposure Goals
OE 3.6, Rev 5	Condition Reports
ON1090.04, Rev 3	Containment Entry
WN0598.076, Rev 0	Moving High Dose Rate Containers (>1R/Hr)

Quality Assurance Reports

Radiation Protection/Process Control/Radwaste Programs Audit, SBK-04-01

Condition Reports

04-01552, 04-03996, 04-03051, 04-03711, 04-01517, 04-01832, 04-03767, 04-01508, 03-11055, 03-09695, 04-01198, 04-01505

Radiation Safety Committee Meeting Minutes

Meeting No. 03-05 dated December 2, 2003
Meeting No. 04-01, dated March 18, 2004

Health Physics Study/Technical Information Document (HPDTID)

Radiological Response to Repair RC-FT-434, (HPSTID 04-001)

LIST OF ACRONYMS

CEVA	Containment Enclosure Ventilation Area
CR	Condition Reports
CBS	Containment Building Spray
CCW	Component Cooling Water
CVCS	Chemical Volume and Control System
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
FSAR	Final Safety Analysis Report
HDR	High Dose Rate
LHRA	Locked High Radiation Areas
MA	Maintenance Manual
MRFF	Maintenance Rule Functional Failure
NCV	Non-cited Violation
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
PI	Performance Indicator
P&ID	Piping and instrumentation drawings
PMT	Post Maintenance Testing
RHR	Residual Heat Removal
RWP	Radiation Work Permit
SFP	Spent Fuel Pool
SI	Safety Injection
TS	Technical Specification
WM	Work Management Manual
WO	Work Order

VHRA	Very High Radiation Area