U. S. NUCLEAR REGULATORY COMMISSION

Region I

Report No. 50-322/82-04

Docket No. 50-322

License No. CPPR-95 Priority - Category B

Licensee: Long Island Lighting Company

175 East Old Country Road

Hicksville, New York 11801

Facility Name: Shoreham Nuclear Power Station

Inspection at: Shoreham, New York

Inspection conducted: February 8-26, 1982

Inspectors:	L H Bettenhausen L. H. Bettenhausen, Ph.D., Chief, Test Programs Section	5/5/82 date signed
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Approved by:	S. D. Ebneter S. D. Ebneter Chief, Engineering Inspection Branch	5/10/82 date signed

Inspection Summary: Inspection on February 8-26, 1982 (Report No. 50-322/82-04)
Areas Inspected: Special team inspection of completed construction of Residual Heat Removal (RHR) and supporting systems ("As-built" Inspection). The inspection involved 373 hours on-site and 73 hours in-office by 3 region-based inspectors, a supervisor and the Senior Resident Inspector.
Results: The RHR and supporting systems generally conformed to approved specifications and drawings. 4 violations and 1 deviation (one-inch HPCI steam drain line with only two check valves for containment isolation, para. 3.4.2; LPCI and RBCLCW do not meet Reg. Guide 1.62 for manual initiation, para. 3.3.3 and 4.2.2, a pipe support did not meet design specifications for alignment, para 4.2.2; housekeeping and fire protection inadequate, paras 4.3.2 and 5.5. Deviation between FSAR description and physical installation for eight specific aspects, para. 3.1.4)

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TABLE OF CONTENTS

DETAILS

1. Persons Contacted

Long Island Lighting Company (LILCO)

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R. DeRocher, Quality Assurance Engineer
D. Durand, Lead Startup Engineer - BOP
* F. Gerecke, QA Manager
W. Hunt, Assistant Construction Manager
W. Klein, Lead Startup Engineer - Electrical
* J. Kelly, Field QA Manager
R. Loper, Technical Support Manager
* J. McCarthy, Section Supervisor - FQA
M. Milligan, Project Engineer
A. Muller, Quality Assurance Engineer, OQA
* M. Museler, Manager, Construction and Engineering
* E. Nicholas, Section Supervisor - FQA
* D. Pluto, Construction Administrator
* M. Pollock, Vice President - Nuclear
J. Rivello, Plant Manager
* J. Rose, Quality Assurance Engineer, OQA
* C. Seaman, Senior Assistant Project Engineer
* J. Smith, Manager Special Projects
D. Terry, Assistant Startup Manager
* E. Youngling, Startup Manager

Stone and Webster Engineering Corporation (S&W)

T. Arrington, Superintendent FQC
* J. Carney, Head of SEO
R. Costa, Project QA Manager
E. Hall, Supervisor - FQC
P. Hawkins, Control Engineer - Instrumentation
R. Morris, Design Engineer
* J. Reiss, Electrical Superintendent

General Electric Company

K. Nicholas, Lead Startup Engineer - NSSS
J. Reilly, Operations Manager

Burns and Roe Corporation

* R. Grunseich, Senior Licensing Engineer

U. S. Nuclear Regulatory Commission

* R. Gallo, Chief, Reactor Project Section 1 A, Region I

* denotes personnel in attendance at the exit meeting of February 26, 1982.

2. Inspection Purpose; Summary of Results

2.1 Purpose and Scope of In... n

The purpose of this inspection was a comparison of the completed construction and physical installation (called the as-built plant) at the Shoreham Nuclear Power Station with regulatory commitments and engineering and design documents. A completed Emergency Core Cooling System and the systems, structures and components required to support its safety functions were selected for inspection.

Team members inspected the physical installation of the Residual Heat Removal (RHR) System and compared the installation to flow diagrams, logic diagrams, construction drawings, and other design and engineering information. Selected portions of other plant systems which are required pport the RHR system in normal and emergency operation were a ected. In the course of the inspection, random sampling was don. . construction and management control activities such as purchase documentation, material control, quality control inspections, repair and rework, engineering and design changes, and maintenance of completed installation.

2.2 Summary of Inspection Results

The RHR System and those portions of support systems inspected were built as described by drawings and specifications, with only minor discrepancies between drawings and piping. The physical installation and its functioning deviated in eight aspects from descriptions in the Final Safety Analysis Reports. The more significant of these were (1) installation of Control Room electrical cabinets in a manner different from that analyzed and described in the FSAR and (2) ventilation duct work blocking some Primary Containment cooling spray nozzles.

Four apparent violations were identified. (1) A one-inch steam drain line was connected directly to the suppression pool containment atmosphere with only two simple check valves outside containment for isolation in violation of General Design Criterion 56 for containment isolation valves. (2) Neither the Low Pressure Coolant Injection and its auxiliary systems nor the Reactor Building Closed Loop Cooling Water system met Regulatory Guide 1.62 requirements for system-level manual initiation. (3) A pipe support was found out of design specification due to inadequate maintenance. (4) Housekeeping and fire protection in diesel generator, fuel oil transfer and screenwell pumphouse rooms were poor. These last two violations were corrected prior to the close of the inspection.

Four observations were made by the inspection team. (1) A large volume of Engineering and Design Change Reports (E&DCR's) were found where timely incorporation of these E&DCR's into drawings and specifications appeared lacking. Although no errors or violations were identified as a result of the practice, the licensee has recognized this as a problem and has initiated a program to reduce the backlog of unincorporated E&DCR's. (2) The issue of electrical separation between cable trays and between Class 1E and non-class 1E electrical cables has not been completely specified. Plans to review and inspect cables for electrical separation were incomplete. (3) Proposed Technical Specifications did not include safety-related, plant-unique systems and did not reflect detail of the completed plant for pipe restraints examined during this inspection. (4) Corrosion of bolts on flanged piping had been documented. A plan of correction was discussed; the corrective action presented did not appear to be thorough and comprehensive.

The inspection report provides details of the physical inspection and the engineering and design information used in the review. The information used is referenced in Section 8 of the report. Discrepancies were discussed with licensee management as they were identified in the course of the inspection and summarized at an exit meeting closing the inspection on February 26, 1982.

3. The Residual Heat Removal System - Comparisons with Codes, Standards, Regulations, Specifications and Drawings

The Residual Heat Removal (RHR) System, designated system E11 at Shoreham, has important operational and safety functions. The physical inspection concentrated on those structures, systems and components whose functions support three modes of operation of the RHR. The three modes are the Low Pressure Coolant Injection (LPCI) mode - a portion of the Emergency Core Cooling System, the Shutdown Cooling mode, and the Suppression Pool Cooling mode. The RHR system is Nuclear Safety Related, QA Category I.

The LPCI subsystem is an integral part of the RHR system. It is designed to restore and maintain coolant inventory in the reactor vessel following a Loss-of-Coolant Accident (LOCA). LPCI is a low head, high flow subsystem delivering coolant from the suppression pool to the reactor vessel. LPCI uses four a-c motor-driven centrifugal pumps in two loops, A and B. The associated valves automatically align the RHR to the LPCI mode when high primary containment pressure or low reactor water level are sensed; the valves isolating RHR from the reactor coolant system are opened when reactor pressure falls below the isolation setpoint. A portion of the flow can also be directed to spray nozzles in the primary containment to reduce temperature and pressure. The RHR system can be aligned to perform shutdown cooling by circulation of reactor coolant from a recirculation loop through one or both RHR loop heat exchangers and then back to the reactor vessel through the recirculation loops. The RHR heat exchangers are cooled by the Service Water System. In the Suppression Pool Cooling mode, the RHR system can be aligned to take water from the suppression pool, pump it through the RHR heat exchanger(s), and return the cooled water to the suppression pool.

The physical inspection compared piping, pipe supports and structures, instrumentation and controls with design drawings, logic diagrams, written descriptions in the Final Safety Analysis Report (FSAR) and Safety Evaluation Report (SER), construction specifications and applicable codes, standards and regulations. The sections which follow describe the various aspects of the inspection - piping and supporting structures, instrumentation and control, electrical wiring, operator control of the system - and the inspection findings. Detailed references, drawings, and documents which were used can be found in Section 8 of this report.

3.1 RHR Piping and Pipe Supports

3.1.1 Discussion

The inspector visually inspected the installed piping and structural supports for the RHR system. The visual inspection consisted of physical verification of piping runs, location, orientation and protective maintenance of supports, hangers, valves, instrumentation taps, insulation, and fittings. The inspection was carried out by tracing the installed piping in the Reactor Building and the primary containment. The inspector compared the installed components to the approved design drawings as modified by Engineering and Design Coordination Reports (E&DCR's) to verify that the as-built configuration of the system agreed with the as-analyzed and approved design. The general workmanship of the installation was also inspected. Additionally, the inspector performed dimensional checks and physical measurements of piping and support structures on a selected basis. These measurements were compared to the detailed isometric drawings of the piping system and pipe supports. In addition, the system flow logic and operational adequacy of the system was evaluated from drawings and the requirements of the design and system description in FSAR and SER. Discrepancies found are discussed in 3.1.4 below.

3.1.2 Pipe Supports

The pipe supports selected for dimensional check were PSSP-807, PSSP-808, and PSSP-819. The measured dimensions of these supports were compared with the 12 applicable drawings to determine their conformance to the design requirements. The inspector also visually inspected several other pipe supports, and reviewed the associated documentation to verify the acceptability of materials, fabrication and installation practices and controls. No discrepancies were identified.

3.1.3 Piping

To determine the conformance of the selected sample of installed piping to the designed and analyzed configuration, the inspector compared the pipe routing and checked dimensions of selected piping bends and elbows, ratings of equipment and orientations of motor operators, valves and other fittings with the 8 applicable isometric drawings. In addition, the inspector also verified that the installed location, orientation, and ratings of shock suppressors (snubbers) on the system conformed to the design. The inspector further observed that the snubbers were well protected by plastic covering and iron protective frames from damage that might result from adjacent construction activities.

The inspector also reviewed the supporting documentation to verify the adequacy of records and held discussions with licensee and architect and engineer (A/E) personnel to determine the adequacy and validity of the approved design of the system in relation to the system description submitted in the Final Safety Analysis Report.

3.1.4 Inspection Findings

Based on inspection, review, and discussions, the inspector determined that the "as-built" configuration of the RHR system piping and appurtenances generally conformed to the approved specifications, drawings, and system description as required by the design.

The inspector, however, identified several areas which apparently deviated from the FSAR description and commitments of the licensee. These deviations are as follows:

-- FSAR Chapter 6.2 and Figure 6.2.5-7 described Primary Containment Spray and number of spray nozzles. The inspector observed that some drywell spray nozzles were blocked by ventilation duct work.

-- FSAR, p.7.3-22 states that valves from other RHR modes are automatically positioned so that water is correctly routed. Contrary to this E11*MOV-055 and 056, one inch RHR Heat Exchanger vents to Suppression Pool, and E11*MOV-057, RHR cooling water to Hydrogen Recombiner, are not automatically positioned.

-- FSAR Fig. 7.3.1-6 and Table 7.3.2-4 shows LPCI Loop selection logic and instruments. Contrary to this, the logic has been deleted.

-- FSAR Table 7.3.4 shows trip set points of 2 psig for high drywell pressure and 500 psig for LPCI low pressure. Page 6.3-12 and Table 6.3.3-6 also give the LPCI low pressure set point of 500 psig. Contrary to this, present setpoints are 1.69 psig and 409 psig, respectively.

-- The following items of FSAR Figure 7.3.1-10A&B were observed by the inspector not to agree with piping drawings and physical inspection:

- Loop fill on B loop should be between valves F015 and F017.

- Relief valves F030A-D go to floor drains, not CRW.

- Relief Valve F025 is not a thermal relief, contrary to Note 12.

- Location of line to Radwaste thru valves MO-F040 and F049 shown incorrectly.

- Cooling water to RHR pumps is RBCLCW, not the emergency equipment cooling water.

- Drains from RHR pump suction and discharge do not tie together.

-- FSAR, p. 5.5-22 states that a relief valve on the RHR pump discharge and another on the RCIC steam supply protect the heat exchanger. The inspector observed that one relief valve was on the discharge line into the heat exchanger, two valves removed from the RHR pump discharge, and the steam supply in from HPCI, rather than RCIC.

-- FSAR, p.7.3-25 states that only the air operated check valve and check bypass valve are located in containment. Contrary to this, a manual isolation valve and manual test, vent and drain valves and connections are located in primary containment.

These items were discussed with licensee staff as they were identified. As a result of the discussions, commitments were given to make appropriate FSAR changes and corrections. Collectively, these items, together with the item identified in Paragraph 5.6 below, constitute a deviation (322/82-04-01).

A number of minor discrepancies between flow diagrams and existing piping and hardware were also identified. They are:

-- FM-20B-13, Note 2, states, "All Motor Operated Valves (MOV's) shall have remote manual switches and indicator lights both local and in Main Control Room". There were no local manual switches or indicator lights for MOV's in system E11;

-- FM-20B-13, Note 3, states, "All MOV's are AC unless otherwise noted". At least 3 MOV's (MOV-51, 53 and 48) are DC and are not so noted.

-- FM-15C-9: TE-020B was physically located on opposite side of valve VGS-60B-3 from that shown.

-- FM-15A-12: Drains from *P-005A and B drawn as going to CRW, but reference locations on Drawing M-10148 are not correct.

-- FM-47A-11: FE-117A and B are not constructed in accordance with Note 15 and no exception is indicated on the drawing.

-- FM-44A-10: No bird screens were present on crankcase vents per the drawing.

-- FM-20 A & B show capped vent and drain lines; most vent and drain lines remain uncapped.

-- FM-20 A & B, among other drawings, show locked valves. No program or hardware is in place to lock valves.

These discrepancies are collectively considered an unresolved item (322/82-04-02).

3.2 Instrumentation, Controls and Electrical Power

3.2.1 Discussion

The LPCI mode obtains safety-related ac and dc electrical power from several sources. Instrumentation and controls are provided for automatic and manual operation. The inspector examined hydraulic and electrical logic and construction, wiring and cabling and a plant-unique valve power system.

3.2.2 Verification of Panels 018 and 021

The inspector observed completed work, partially completed work and reviewed quality records documenting work performance. The inspector examined panels and traced instrument lines from panels E11*PNL-018 and E11*PNL-021 to the root tap on the RHR system piping as follows:

From Panel E11*PNL-021

-- Tap No. A-3, line 1E11 *½K 1014-1C-N9-2.

-- Tap No. A-6, line 1E11 *½K-1010-1C-N9-2.

-- Tap No. A-7, line 1E11 *½K-1011-1C-N9-2.

-- Tap No. A-8, line 1E11 *½K-1012-1C-N9-2.

-- Tap No. A-10, line 1E11 *½K 1013-1C-N9-2.

-- Tap No. A-11, line 1E11 *½K-1015-1C-N9-2.

From Panel E11*PNL-018

-- Tap No. A-3, line 1E11 *½K-1009-1C-N9-2.

-- Tap No. A-6, line 1E11 *½K-1006-1C-N9-2.

-- Tap No. A-7, line 1E11 *½K-1004-1C-N9-2.

-- Tap No. A-8, line 1E11 *½K-1005-1C-N9-2.

-- Tap No. A-10, line 1E11 *½K-1007-1C-N9-2.

The inspector noted that the metal identification tags were missing from instrument line No. ½K1007 at the instrument panel and from instrument line nos. ½K1004 and ½K1005 at the root valve. In addition, the inspector observed that a number of vent valves had not been plugged or capped to prevent dirt and dust from entering the valves mounted in panel nos. E11*PNL-021 and E11*PNL-018. The licensee took immediate corrective action to replace the missing tags and to cap the exposed valve openings.

3.2.3 Verification of Instrument Line Routing

Using six different weld map drawings, the inspector verified weld location and type of weld, couplings and fittings used in routing instrument lines from the instrument panel to the process line and/or instrument.

The inspector noted that the weld map drawings were identified as "as-built" drawings. The inspector questioned the lack of information on existing drawings regarding instrument elevation. The licensee stated that the weld maps were the only "as-built" requirements of the ASME code. However, the inspector emphasized the importance of knowing the elevation at which the particular instrument was located so that an operator might know which instruments may be lost in the event of flooding as has been demonstrated by the accident at TMI. The licensee agreed to include information on instrument elevation on FK-1AA instrument location drawings for Elevation 8.

3.2.4 Electrical Logic and Wiring

Using electrical block diagrams, SK logic diagrams, test loop diagrams and cable pull tickets, the inspector verified instrument electrical functions, cable routing and terminations for the RHR system. This verification included "A" and "B" RHR system flow, "A" and "B" RHR Heat Exchanger level and "A" and "B" RHR Heat Exchanger Level Controller Output.

The inspector observed that several electrical jumpers used in Control Room panel 612 and two wires from cable 1B21BBX198 in Control Room panel 601 which had been removed from the terminal block were not tagged. The free terminals were not protected against possible shorting of adjacent terminations. The licensee took immediate corrective action by tagging the jumpers in panel 612 and reconnecting the two-conductor cable of panel 601. Personnel were re-instructed on the requirement for tagging all jumpers and for providing protective cover for exposed wire leads. The inspector had no further questions in this area.

3.2.5 Swing Bus Design

To meet 10 CFR 50, Appendix K, requirements for a recirculation system line break and to assure that redundant power systems for the valve buses are independent, LPCI valve swing buses were provided such that a single failure of the valve power system will not jeopardize any emergency bus. The emergency power system supplying the valves is designed to comply with Regulatory Guide 1.6 and IEEE-308-1974. The design uses four Class-1E motor-generator (M-G) sets as isolation devices, and supplies independent power to two valve buses (FSAR Figure 8.3.1-10). Loss of the normal power source (including failure of an M-G set) will not affect operation of the valves since the affected valve bus will automatically transfer to the alternate power source. A failure of a valve bus (if not cleared by the class 1E breakers at the M-G output) will not trip the main source breaker, since fault current is not passed back through the M-G set feeder breaker. Two M-G sets, Nos. R24*MG-111 and R24*MG-113A, are energized from diesel generators 101 and 103, respectively, and supply power to valve bus No. R24*MCC-111X. The remaining two units, Nos. R24*MG-112 and R24*MG-113B, are energized from diesel generator 102 and 103, respectively, and supply power to valve bus No. R24*MCC-112Y.

Power and control cables associated with the M-G set motor feeder breakers are Class 1E. Power and control circuits associated with each valve bus and downstream of the M-G sets are indeoendent and are run in rigid conduit with the required separation between each of the two valve bus systems and between those systems and the three emergency electrical onsite power systems. Control devices and wiring associated with equipment downstream of the M-G sets are isolated by metal barriers from all other wiring within the control and relay panels.

The inspector verified that the automatic transfer scheme complies with RG 1.6 and IEEE 308-1974 by review of wiring diagrams and plant observations.

The inspector examined equipment located in Elevation 150 switch gear rooms and traced conduit/cable routing through the use of cable pull tickets and conduit routing cards. The inspector verified panel and equipment terminations for switchgear Nos. 1R23*SWG-111, 1R23*SWG-113 and 1R23*SWG-112; Control panel Nos. 1R24*PNL-111, 1R24*PNL-112, 1R24*PNL-113A and 1R24*PNL-113B; electrical interlock nos. 1R24*TRS-11X and 1R24*TRS-112Y; and motor control center Nos. 1R24*MCC-111X and 1R24*MCC-112Y.

The inspector traced the conduit and cable routing from the motor control centers, through penetration Nos. WB3 and EB1 to the recirculation pump discharge motor operated valve Nos. B31*MOV-032A and B31*MOV-032B, respectively. In addition, the inspector traced the conduit/cable routing from the motor control center to the RHR outboard motor operated valve Nos. E11*MOV-036A and E11*MOV-036B and the inboard motor operator valve Nos. E11*MOV-037A and E11*MOV-037B. Cable routing and verification of terminations included cables from the M-G sets to the transfer switch panel and the switchgear panels.

During this inspection, the inspector observed several apparent violations of separation criteria between non-class 1E cable and Class 1E cable and one violation between Class 1E cables of different divisions. This observation is discussed in the section which follows.

3.2.6 Electrical Cable Separation

The inspector noted that separation of cables in transition from tray to tray and tray to conduit was not addressed specifically in the FSAR nor was transition separation addressed in electrical installation Specification SH1-159. Resolution for E&DCR No. F-27961 dated August 8, 1980, imposed the same separation criteria for cable in "free air" as for trays and conduits. The licensee indicated that these requirements were imposed after the installation of a majority of the cables. The inspector observed that the required one-inch horizontal and one-foot vertical separation criteria were not maintained for non-class 1E/class 1E cables in transition from one raceway to another for the following raceway groupings:

-- 1TC616N, 1TC606N, 1TK616N, 1TK616R, 1TK605R, 1CX605SNA and 1CK605RA.

-- 1TC400R, 1TC404R and 1TC411N

-- 1TK794B, 1TK785N

The required separation between Class 1E cables of different divisions was not maintained for this raceway grouping:

-- 1TX706N, 1CC706ND, 1CC705RR, and 1CC705BL.

E&DCR F-27961C was issued on February 23, 1982 tc document the separation violation of 1TK794B/1TK785N.

The licensee stated that a walkdown was planned for all safety related cable as part of the electrical "as-built" program and that the separation requirements for "free air" cable would be verified at that time. However, the licensee was not able to provide written instructions or procedures for this planned verification.

Additionally, the inspector noted that the FSAR method for determining separation did not agree with the definition given in IEEE 384-1974. FSAR Section 3.12.3.5.2 states, in part: "... vertical separation is measured from the bottom of top tray to bottom of the side rail of bottom tray". The IEEE 384-1974 definition (page 11) states, in part: "... vertical separation is measured from the bottom of the top tray to the top of the side rail of the bottom tray". The licensee's method provides 8 to 9 inches between trays versus the 12 inches specified by IEEE-384-1974. In addition, NRC question 223.12 asked the licensee to compare the FSAR separation requirements to those of IEEE-384-1974 and RG 1.75 and to discuss the reasons for concluding that the less stringent criteria are adequate. The licensee response to question 223.12 did not address this difference between the two documents. This question will receive further NRC review.

The electrical separation difficulties at Shoreham date back to 1978. E&DCR F-13072 issued May 1, 1978, stated that separation criteria for conduits could not be met and requested approval of a nonconforming installation. E&DCR F-19039 issued March 14, 1979 permitted installation of cable into raceways known to be in violation of the separation criteria, defined in FSAR sections 3.12.3.5.2.C and 3.12.3.5.2.d, provided that it was documented on an E&DCR (NRC Inspection Report 322/79-0,. Licensee response to the item of noncompliance (322/79-07-02) indicated that full compliance, including final disposition of all E&DCR's and completion of any necessary rework, would follow completion of cable installation at the site. The inspector reviewed two recent

E&DCR's, Nos. F-39477 and F-39480, in draft form which indicated that the disposition of E&DCR's for electrical separation was in progress. The issue of electrical separation is assessed as a weakness and is assigned Item No. (322/82-04-03).

3.2.7 Conduit Sealing

During RHR system walk-downs, the inspector nr ed that electrical components were not completely sealed to prevent moisture entry. The final run of cable to a component was often via a metal conduit. The conduit was not sealed where the electrical cable entered. This opening provides a moisture entry path to the component.

A review of the associated documentation revealed that conduit sealing was required. Two Valcor Engineering Corp. solenoid valves in the RHR system (E11*SOV-166A and 167A) had the following note in the manufacturer's technical manual: "Owner is responsible for sealing the conduit connection and preventing the entrance of moisture thru the conduit to maintain the validity of the IEEE-323 qualification". Also, E&DCRs F-5750 and F-5750A, dated December 7, 1976 and January 6, 1977 respectively, stated that the Reactor Building and other areas are considered wet locations and required that conduits in these areas be sealed.

Despite the above requirements, the licensee was unable to identify any existing program or procedure which would seal the subject conduits or inspect the adequacy of the seals, once installed. This item is unresolved and is designated Item No. (322/82-04-04).

3.2.8 IE Information Notice 81-01:Possible Failures of General Electric HFA Relays

General Electric Service Advice Letter (SAL) 721-PSM-152.2 explains that the Lexan coil spools on HFA relays are subject to cracks which might prevent desired contact action. The licensee stated that all HFA coils used in NSSS systems have been replaced. The HFA coils in Balance of Plant Equipment (Cat I & II) will be replaced by April, 1982. The inspector had no further questions in this area.

3.3 RHR System Controls

3.3.1 Discussion

The inspector reviewed the RHR system controls including: automatic and manual initiation circuitry, reset circuitry, selected pump and valve logic, control room switches, indicators, labels, and mimics, remote shutdown panel controls, and selected local instrumentation.

The controls were reviewed against applicable regulatory requirements, licensee commitments, good human factors practices and inspector judgement. With the exception of the items below, no discrepancies were identified.

3.3.2 Labeling

The inspector noted the following RHR system labeling deficiencies:

-- Annunciator 1122 has a seemingly contradictory label.

-- The mimic for E11*MOV-50 and B-loop drywell spray is incorrect in the control room and the remote shutdown panel.

-- The mimic for lines through E11*PCV-007B is incorrect.

-- The temperature points on the E41-TR100 (HPCI and RHR temperature recorder) are labelled only with General Electric numbers, not LILCO identifying numbers. This is also true for other recorders.

-- The different points on E41-TR100 (a 24 point recorder) do not have a cross-reference between colors and numbers on the label for easy identification. This is also true for other recorders.

-- The label on the Shutdown Cooling Isolation Reset Button for E11*MOV-037 is confusing.

-- E11*SOV-061 and 062 in the control room actually control AOV's but this is not indicated on the control room labels.

-- The controllers for E11*PCV-003B and E11*PCV-007B are not labeled as such.

-- Local instruments are not clearly labeled as to function.

These items had not been specifically addressed in earlier control room human factors reviews.

These items are unresolved and are collectively designated Item No. (322/82-04-05).

3.3.3 Manual Initiation

Review of the manual initiation capability provided for the LPCI mode of the RHR system revealed that the licensee's system is not adequate. The regulatory requirement for manual initiation originates in 10 CFR 50.55a(h) which requires that protection systems meet IEEE-279-1971.

The LPCI mode of RHR is a protection system as defined in IEEE-279. Paragraph 4.17 of the standard requires that the protection systems include means for manual initiation of each protective action at the system level. Regulatory Guide 1.62, "Manual Initiation of Protective Actions", (RG 1.62) describes an acceptable method for complying with Section 4.17 of IEEE-279-1971. Paragraph 7.3.2.1.2.19 of the Shoreham FSAR states that the Emergency Core Cooling System (including LPCI) meets RG 1.62. Paragraph C.2 of RG 1.62 states that manual initiation of a protective actior at the system level should perform all actions performed by automatic initiation, such as starting auxiliary or supporting systems and sending signals to appropriate valve-actuating mechanisms to assure correct valve position.

The LPCI manual initiation switch does not provide signals to place the following auxiliary or supporting systems in the accident mode: RBCLCW for the RHR pump seals, area coolers for the RHR pump motors, or chilled water to RHR area coolers. Additionally, the following eight LPCI valves are not sent signals to assure correct valve position from the manual initiation circuitry: E11*MOV-051, 052, 053 and 054; E11*AOV-061A & B; and E11*AOV-062A & B. The inspector noted that under certain conditions, the features provided with the manual initiation switch would be sufficient to manually initiate LPCI, but that under worst case assumptions this would not be true.

The condition described above was one instance where measures established by the licensee did not assure that applicable regulatory requirements, as specified in the license application, were correctly translated into drawings in accordance with 10 CFR 50, Appendix, B, Criterion III. Another instance is described in paragraph 4.2.4 for the RBCLCW system. This item is a violation and is designated as Item No. (322/82-04-06).

3.3.4 Override or Bypass indication

IE Circular No. 78-19, "Manual Override (Bypass) of Safety System Actuation Signals" describes a situation where automatic safety functions unintentionally were made inoperable and the condition was not indicated in the control room. The inspector reviewed this Circular and the licensee's response as they related to the RHR system. The licensee has a sophisticated monitoring system to annunciate these conditions. However, one problem area was noted.

The licensee's internal response to Circular 78-19 refers to the response to IE Bulletin 79-08, (events relevant tc BWR's identified during Three Mile Island incident) item 6, which discusses controls for valve positioning.

The Bulletin 79-08 response states that, if a motor operated valve (MOV) has a given safety position and it is moved from that position with consequent loss of ability to return automatically, then its respective system "inop" alarm is sounded.

Two areas of the RHR system were noted not to satisfy this commitment:

- closure of a single RHR pump suction valve, E11*MOV-031; and
- the case where E11*MOV-037 A and B are blocked closed by a shutdown cooling isolation signal.

The inspector noted that closure of two RHR pump suction valves in a loop would give the system inop alarm. However, closure of a single suction valve renders the loop inoperable because the Shoreham Draft Technical Specifications require that both RHR pumps in a loop be operable.

The inspector noted that the blocking of E11*MOV-037A and B closed was of particular concern to the LPCI function in the Shutdown Cooling Mode. E11*MOV-037A and B are the LPCI loop injection valves for loops A and B. If Loss of Coolant Accident (LOCA) occurs while in the Shutdown Cooling Mode, reactor vessel level will decrease and the shutdown cooling valves will close. E11*MOV-037A and B will close also. As reactor vessel level drops further to the ECCS initiation setpoint, LPCI will be automatically initiated. However, neither E11*MOV-037A or B will open, since the logic blocks them closed until the Shutdown Cooling Isolation has been reset by the control room operator. This closure block is considered significant enough to warrant incorporation into the system "inop" alarm, and in fact is committed to in the licensee's response to Bulletin 79-08. The issue of fully meeting commitments of the Bulletin 79-08 response is unresolved and is designated Item No. (322/82-04-07).

3.3.5 Remote Shutdown Panel

The purpose of the Remote Shutdown Panel is to provide a system outside the main control room to bring the reactor to a cold shutdown condition. The Panel does this irrespective of shorts, opens or grounds in the control circuits in the main control room resulting from an event that necessitated evacuation of the control room.

The Remote Shutdown Panel C61*PNL-001, controls various components of the Nuclear Boiler System, Reactor Recirculation System, Residual Heat Removal System, Reactor Core Isolation Cooling System, Fuel Pool Cooling System, Service Water System and the Reactor Building Closed Loop Cooling Water Systems. Normal reactor cooldown is accomplished by controlling the various system components from the control room. Cooldown can be accomplished from the Remote Shutdown Panel when feedwater is unavailable and when the reactor is isolated from its normal heat sink. The Remote Shutdown Panel was inspected for conformance to drawings and specifications including electrical workmanship, separation and the human factors and labeling discussed in Section 3.3.2.

The inspector observed completed and partially completed work to determine whether it was accomplished in accordance with applicable specifications, NRC requirements and licensee commitments in the areas of installation, routing, separation and terminations. The inspector noted that changes and additions to wiring and logic diagrams to reflect the "as-built" condition were being made. In addition, the licensee indicated that scheduled modifications were physically complete except for minor modification to RCIC power supply and the LPCI annuciators.

The inspector had no further questions concerning the Remote Shutdown Panel.

3.4 Containment Isolation Valves (CIV's)

3.4.1 Discussion

The RHR System penetrates the primary containment in a number of places. The piping to each of these penetrations has containment isolation valves for isolating the lines under accident conditions. The inspector reviewed various design and test documents and observed the RHR System CIV's in the plant for the following:

-- Proper valve type, location, and arrangement.

-- Adequate valve stroke time testing and leak rate testing.

-- Adequate description in the proposed Technical Specifications.

-- Proper physical condition and protection from damage.

With the exception of the items in the three paragraphs below, no new discrepancies were identified. The inspector did note that the licensee had not yet resolved a previous violation (322/81-02-01). This violation cited a situation where CIV's were not located as close as practical to containment. Some RHR system CIV's are located similarly.

3.4.2 General Design Criterion 56

During the review of the containment isolation valve designs for the RHR system, the inspector identified one line whose valves did not meet the requirements of 10 CFR 50, Appendix A, Criterion 56. This criterion describes the CIV's required for lines which penetrate the primary containment and connect directly to the containment atmosphere. The RHR system line connected to penetrations X43 and XS-5 is such a line. For this line Criterion 56 requires two CIV's, which must be either automatic or locked closed and which must not be check valves. A HPCI steam drain line ties into this RHR system line and has only two check valves (numbers 3144 and 3145) as containment isolation valves. The arrangement is depicted in FSAR Fig. 6.2.4-2. This violation of GDC-56 was not identified nor justified prior to the inspection. 10 CFR 50, Appendix B, Criterion III requires correct translation of applicable regulatory requirements. This item is a violation and is designated as Item No. (322/82-04-08).

3.4.3 Leak Rate Testing

The inspector reviewed portions of the draft procedure for performing Type C leak rate testing on containment isolation valves and discussed test methods with the cognizant Startup Test Engineer. The inspector reviewed the testing proposed for RHR system CIV's in order to verify that it would be in accordance with 10 CFR 50, Appendix J, and that tests would conservatively measure CIV leak rates. Testing plans generally met these conditions. The inspector had additional questions in two areas.

3.4.3.1 Reverse Direction Testing

The first area was the testing planned for globe valves tested with the pressure applied to the valve in the reverse direction from that which the valve would experience during a LOCA. For these, the licensee had stated in a submittal that the testing would be conservative since the valves were normally seated with a force at least three time greater than the test pressure seating force. The inspector reviewed data from the valve vendors and from valve testing on site to verify that this was accurate. The "three times" criterion was met for all valves except two, E41*MOV-049 and E51*MOV-049, as purchased.

These two valves required additional demonstration of meeting the "three times" criteria; this was provided by the vendor. The inspector questioned controls existing to ensure that closing force for these two valves would be maintained throughout plant life, including maintenance or replacement. For maintenance, the proper valve actuator touque switch settings are documented. For replacement, the licensee acknowledged that existing controls might not be sufficient. Therefore, prior to completion of the inspection, the licensee issued an E&DCR to the valve purchase and specification, which noted the "three times" requirement for these two valves to ensure proper controls if replacement is required. The inspector had no further questions in this area.

3.4.3.2 Downstream Venting

During review of Type C leak rate test procedure PT.654.003, the inspector noted that not all valve tests ensured that the post accident differential pressure (Pa) would be applied across the valve under test, since the tests did not always provide an atmospheric vent path downstream of the valve under test. If the valve under test leaked significantly, the volume downstream of that valve could pressurize, thus reducing the differential pressure across the valve. This situation would give artificially low test results. Some valves for which downstream venting was not specified are:

- penetration X10A: valves G11*MOV-639 and the G11 check valves.

- penetrations X-42/XS-5: valves E11*01V-3144, MOV-55A & B, and MOV-56A&B.

- penetrations X-8A&B: valves E11*MOV-042A&B.

The inspector noted that the procedure did not ensure that proper downstream venting was provided for each valve test. This is unresolved and is designated as Item No. (322/82-04-09).

3.4.4 CIV Timing

FSAR Table 6.2.4-1 specifies maximum CIV closure times. Each CIV is tested and timed after final installation, using Checkout & Initial Operation (C&IO) procedures. These procedures specify acceptance criteria for CIV closure times. The actual opening and closing times are recorded. The licensee also has specified required CIV closure times in the proposed Technical Specifications. The inspector reviewed these documents and noted that the times being used were not consistent. The licensee stated that reanalysis had changed a number of CIV closure times and that an FSAR change was being processed to revise Table 6.2.4-1. Additionally, the licensee stated that any C&IO tests which had not been done to the latest criteria would be redone if necessary. The inspector reviewed internal memoranda documenting the above and had no further questions at this time.

4. Support Systems

Support systems are those systems in use or ready to be used to support the RHR System in its modes of operation. Support systems include the Service Water (SW) System (P41); the Reactor Building Closed Loop Cooling Water (RBCLCW) system (P42); the Emergency Diesel Generators (EDG), including fuel oil storage and transfer systems, air start system and service water for cooling; the ECCS discharge line fill system; and the Leakage Return System.

4.1 Service Water System

4.1.1 Description

During normal operation, the SW system provides cooling water to the RBCLCW heat exchangers, the drywell cooling booster heat exchangers, the turbine building closed loop cooling water heat exchangers, the reactor building and control room air conditioning chilled water condensers, the main chilled water condensers, and other nonsafety-related components. The service water system is also designed to provide cooling water to the RHR heat exchangers to remove reactor decay heat during a scheduled shutdown or accident conditions. The system also provides cooling water to the EDG engine coolers, emergency makeup water to the spent fuel pool, and emergency cooling water to the ultimate cooling connection.

4.1.2 Physical Inspection

The inspector verified that the service water system conformed to the approved final design. Pumps, heat exchangers, piping, instrumentation, valves, supports and restraints were inspected by direct observation.

The physical installation agreed with piping and instrumentation diagrams, including those contained in the FSAR. The inspector verified that the system agreed with the FSAR descriptions.

4.1.3 Corrosion of Carbon Steel Bolts

During inspection of the SW system, the inspector observed that carbon steel bolts and nuts which hold together the copper-nickel (Cu-Ni) flanges of the service water piping had corroded. Salt water and two dissimilar metals in contact caused corrosion of the bolts and nuts by electrolysis and galvanic corrosion. The inspector reviewed licensee actions to replace corroded bolts and to prevent recurrence.

The licensees' representative stated that, prior to ASME certification of the system, plastic insulation kits would be installed on the bolts and nuts to separate them from the Cu-Ni flanges. The inspector expressed concern that only bolts and nuts corroded substantially would be replaced and that this might be done on selected flanges only. The licensee's representative stated that they were aware of the corrosion problem on the service water system, that the system had not been ASME certified and that bolts and nuts on the flanges were temporary. The problem of bolt corrosion would be resolved finally upon ASME certification of the system.

The inspector expressed cor-ern that there was not an adequate program to identify and replace all corroded carbon steel bolts and nuts on all Cu-Ni flanges of the service water system, the corrective action taken to date has not involved appropriate levels of management, and that the problem may not have been thoroughly reviewed for reportability to NRC. The issue of corroded bolting on Cu-Ni piping is assessed as a weakness and is assigned Item No. (322/82-04-10).

4.1.4 Biofouling in Salt Service Water (IE Bulletin 81-03; IE Information Notice 81-21)

IE Bulletin 81-03 pertains to bio-fouling and clogging of salt water service systems supplying safety related systems. In its original response, the licensee stated that biofouling had taken place in the non-safety related Turbine Building Service Water System. The blue mussel (Mytilus edilus) was found in the 24-inch supply pipe to the Turbine Building Closed Loop Cooling Water System Heat Exchanger (TBCLCW). The Reactor Building Closed Loop Cooling Water Heat Exchangers were also inspected and found to be free of bio-fouling.

To assure adequate flow, the licensee has indicated that all heat exchangers in safety-related systems using service water will have flow elements either on inlet or discharge and that these heat exchangers will be monitored for flow during plant operations. The licensee has agreed to provide a revised response to Bulletin 81-03 with details of the monitoring program. This item remains open pending NRC review of the additional licensee submittal. Information Notice 81-21 pertains to RHR baffle deformation induced by high differential pressure resulting from blockage. The licensee expects to complete its engineering evaluation by April 1, 1982. Preliminary recommendations call for pipe line insert strainers at the inlets to the RHR exchangers and monitoring of the differential pressure across these strainers. A rise in differential pressure across the strainer evaluated in conjunction with normal monitoring of the RHR inlet flow would enable early detection of potential blockage. The inspector noted that the licensee has taken the initiative and is employing engineering effort in an area of potential concern.

4.2 Reactor Building Closed Loop Cooling Water (RBCLCW)

4.2.1 Description

The RBCLCW System provides cooling water to a number of plant systems and components. It is cooled, in turn, by the Service Water System. The RBCLCW system normally operates to supply cooling to both safety-related and nonsafety-related components. Upon an accident signal, the nonsafety-related portions of the system are isolated and the system realigns into the accident mode. RBCLCW is required during an accident (FSAR, p. 9.2-9) to supply cooling water to the RHR pump seal coolers. The inspector reviewed various aspects of the RBCLCW system to verify conformance to regulatory requirements and licensee commitments.

4.2.2 Pipe Supports and Restraints

The inspector visually inspected pipe supports and restraints on RBCLCW on a random basis for obvious defects and workmanship. During the inspection, it appeared that support No. 1P42-PSST-056 was not properly aligned to its vertical axis. The inspector verified this discrepancy by physical measurements in the presence of licensee representatives and found that the support was $5\frac{1}{2}^{o}$ out of vertical. A tolerance of 4^{o} from the vertical axis was allowed by the specification. The inspector further investigated the cause of this discrepancy by reviewing the Quality Control inspection package and associated documentation for the support. From this review of documents, discussions with licensee engineers, and personal observations of construction activities in the vicinity of the support, the inspector concluded that this discrepancy in the hanger alignment was a result of improper erection of scaffolding in the vicinity of support after the final QC inspection was completed and the support had been accepted. Upon the identification of the discrepancy, the licensee initiated prompt corrective action to restore the support to its design configuration. The inspector examined the support on February 26, 1982 after the corrective action and found it restored to its design configuration. The inspector informed the licensee that failure to maintain the support in acceptable condition was a violation of 10 CFR 50, Appendix B, Criterion II, (322/82-04-11).

4.2.3 Piping

The inspector reviewed pertinent documents and drawings and performed detailed system walk-downs to verify that the RBCLCW system was constructed in accordance with P&IDs and the FSAR.

Two discrepancies were found:

- P42-TE-020B on FM-15C-9 was physically located on the opposite side of valve VGS-60B-3 from that shown on the drawing.

- The drains from P42*P-005A & B are illustrated as going to the Clean Radwaste System which is on Dwg. No. M-10148. However, the indicated reference locations for DWG No. M-10148 were not correct.

This item is unresolved and is another example of the drawing discrepancies discussed in paragraph 3.1.3. This is part of Item No. (322/82-04-02).

4.2.4 RBCLCW System Controls

The inspector performed reviews of the RBCLCW System controls like the reviews of the RHR System controls described in paragraph 3.3.1. With the exception of the items in the two subparagraphs below, no discrepancies were identified.

4.2.4.1 Labeling

The inspector noted the following two types of control room labels did not provide clear indication of their use:

- The RBCLCW valves to and from the recirculation pump coolers are not labeled to show which pump or loop they supply.

- The RBCLCW Heat Exchanger inlet valves are not labeled to show clearly which Heat Exchanger they supply.

4.2.4.2 Manual Initiation

As described in paragraph 3.3.3, the regulations require that the plant protection system include means for manual initiation of each protective action at the system level. Further, FSAR paragraph 7.6.2.5.2.12 states that the RBCLCW system has the required manual initiation features described in Regulatory Guide 1.62. Inspector review of logic circuitry revealed that there was no manual initiation feature at the system level for the RBCLCW System. This item is another example of the failure of design control measures described in paragraph 3.3.3 and is included in part of the Violation, Item No. (322/82-04-06).

4.3 Emergency Diesel Generators

Three fast-starting, onsite emergency diesel generators (EDG) are arranged so that any two can provide necessary power for operation of engineered safety features to assure safe shutdown if offsite power is lost. The EDG's are automatically started on loss of voltage to the generator's 4160 volt bus, high drywell pressure, and low reactor vessel level. If the preferred (offsite) power source is not available, the EDG's are automatically connected to the 4160 volt emergency buses and sequentially loaded.

4.3.1 Emergency Diesel Engine Modifications

The inspector held discussions with startup personnel, reviewed the EDG vendor manuals, E&DCR's, repair and rework requests, P&ID's, and drawings concerning these modifications to the three emergency diesel generators:

-- The engine piston modification consisting of changing the piston crown to piston skirt bolting assembly, including bolts, washers and machining of surfaces.

-- Installation of new expansion joints for the turbocharger on each engine.

-- Installation of a new vibration support for the turbocharger on each engine.

The inspector verified by visual observation, discussions, and review of documentation that the internal and external modifications to the emergency diesel generators would not affect their reliability or operation.

4.3.2 Emergency Diesel Generator Support Systems

Some systems supporting the operation of the emergency diesel generators are the EDG portions of the service water system, the EDG air start system, and the fuel oil storage and transfer systems. These were inspected as discussed below.

4.3.2.1 EDG Service Water System

Service water flows through the diesel engine coolers and, in turn, cools the diesel engine jacket water. The jacket water system removes heat from the diesel engine components during operation. The inspector verified by physical inspection of the EDG service water piping that the physical installation is in agreement with selected isometrics, approved E&DCR's, FSAR description and P&ID's. During this, the inspector observed salt encrustation at all flanges and at top and bottom caps of relief valves P41-ROV-019A and 019B installed on the bypass line of the service water outlet from the diesel engine coolers. This condition usually indicates encrustation inside the valves, as well. If this condition exists, it can make the valves inoperable. The proper operation of these relief valves are subject to further inspection.

4.3.2.2 EDG Air Start System

Each EDG is provided with two independent, redundant air start systems capable of starting the diesel engine without external power. Each air start system has sufficient volume to crank the engine for a minimum of five starts without recharging the tanks. Each motor-driven air compressor has the capacity to recharge the air storage system in thirty minutes to provide the minimum five starts. The inspector verified by physical inspection of the EDG air start system that the installation of the system was in agreement with the P&ID, approved E&DCR's, and the FSAR. No discrepancies were noted.

4.3.2.3. Pipe Supports and Restraints in EDG Rooms

The inspector examined the pipe supports and restraints and equipment support structures in the EDG rooms. A physical dimensional check was also performed on pipe support PSR-169 to determine its conformance to drawing BZ-537-11-58-1. Additionally, the inspector reviewed the supporting documentation packages for supports PSST-10, PSST-12, PSST-13, and PSST-15 for compliance to the requirements for materials, welding, installation process and final QC inspection. The inspector visually examined the structural supports for air handling and fuel systems. These supports were inspected for any obvious defect, and workmanship, proper foundation/ baseplate support, and protection from internally generated missiles. No discrepancies were identified.

4.3.2.4 Fuel Oil Storage and Transfer Systems

The Fuel Oil Storage and Transfer (FOS&T) systems consist of the auxiliary boiler fuel oil transfer system, the EDG fuel oil storage and transfer system, and the diesel engine fuel oil system. Each of the three emergency diesel engines is supplied by a separate FOS&T system to allow seven days continuous operation at rated load. The systems are designed to transfer fuel oil from the auxiliary boiler fuel oil storage tanks to the fill piping for the EDG oil storage tanks. Auxiliary boiler fuel is compatible with diesel engine fuel and can be used for long term operation of the EDG's. Each EDG draws fuel from its own day tank, supplies the needs of the engine and returns the excess fuel back to its day tank. Fuel oil pumps in the transfer system automatically move oil from storage tanks to the day tanks of each diesel engine, as needed, in order to keep the day tank full.

The inspector verified by physical inspections of systems and components, review of vendors' manuals, P&ID's, FSAR descriptions, approved E&DCR's, R/RR's, and discussions with licensee representatives that the physical installation of these systems conformed to the approved final design. The inspector had no further questions regarding the FOS&T System.

4.3.2.5 Physical Structures and Surrounding Areas

The inspector observed the fuel oil filling station in the station yard, the FOS&T system rooms, the EDG rooms, and the surrounding yard area. These observations are described below.

-- Yard Filling Station

A visual inspection of the auxiliary boiler and EDG fuel oil filling station included the filling connections, valves, piping, caps, locking chains, filters, flow meters and auxiliary boiler fuel oil storage tank vents and vent screens. No discrepancies were noted.

-- Fuel Oil Storage and Transfer Rooms

The Mechanical and electrical equipment of all three rooms was inspected. Work in progress observed. The inspector noted the following: (1) the three room vents (from each room to atmosphere) were taped closed. (2) the vent pipe for room A had no screen on it. (3) energized temporary electrical cables were hung from the vent pipes on the roof of rooms A and C. (4) drain pans and drip trays underneath each set of fuel oil transfer pumps, buckets, and drain wells in the corners of the rooms, had fuel oil in them. (5) the transfer pumps leaked when running. These observations were identified to licensee representatives on February 12, 1982. Followup inspections were performed February 23-25, 1982. Transfer pump suction check valve modifications were going on at this time; welding operations were taking place on the check valves. The fire hazards noted above were still present.

The inspector verified documentation for the check valve modification on E&DCR's and supporting diagrams.

-- Emergency Diesel Generator Rooms

The EDG rooms were inspected on several occasions. The inspector observed work in progress, housekeeping, cleanliness, fire protection and fire hazards. In EDG room A, the inspector observed two buckets of fuel oil placed on top of the engine walkway and a five gallon bucket half full of fuel oil sitting on the grating deck over the engine. In EDG room C, the fuel oil day tank was overflowing through temporary plastic hoses on top of the day tank. The hose ran into a bucket placed on the side of the day tank. The bucket was full and was overflowing into a second bucket placed on the grating deck on top of the engine. This bucket was also full and had begun to overflow onto the deck. At the same time, EDG B was being run to test the newly-installed turbocharger vibration support.

Small pockets of fuel oil were observed at the ends of all three emergency diesel engines, while debris, metal shavings, boards and fuel oil had accumulated under the generators at the rear ends of each EDG. The inspector observed welding in progress in EDG rooms A and C while these conditions existed.

-- Yard Area Around FOS&T & EDG Rooms

An inspection was made of the auxiliary boiler fuel oil system rooms underground, just outside of the FOS&T rooms. The inspector examined the fuel oil transfer lines, valves, pumps, piping and instrumentation. No discrepancies were noted.

An inspection also was made of the EDG fuel oil day tank vents and the EDG crankcase vents that extend out of the EDG rooms. The EDG fuel oil day tank vents had flame arrestors installed. None of the EDG crankcase vents had bird screens installed on them as indicated on flow diagram FM 44A-10. The bird screens are to prevent clogging of the crankcase vent line which could result in crankcase explosion. This discrepancy is part of unresolved Item No. (322/82-04-02).

4.3.2.6 Cleanliness and Fire Prevention

The conditions of the EDG fuel oil storage and transfer rooms and the EDG rooms described above constitute a violation of established practices and procedures to prevent fire and maintain cleanliness. Stone & Webster Engineering Corporation Construction Site Instruction 13.1, states, in part, "Work areas shall be kept sufficiently clean and orderly so that construction activity can proceed in an efficient manner ... excess material shall not be allowed to accumulate and create conditions that will adversely affect quality ... Equipment and instructions for the protection from the prevention of damage by fire shall be provided ..."

-- On February 11-12, 1982, and again on February 24, 1982, the following fire hazards were identified in the EDG fuel oil storage and transfer rooms: Fuel oil leaking from pumps; fuel oil in drip trays, wells and buckets; fuel oil fumes in rooms while transfer pumps were running; room vents taped closed.

-- On February 25, 1982, welding of the fuel oil transfer pump suction check valves in room "C" was observed with no fire extinguishers present, no fire watch designated and no cleanup of hazards that were identified on February 24, 1982.

-- On February 11-12, 1982, and again on February 24, 1982, these fire hazards were identified in the EDG generator rooms: Fuel oil overflowing from plastic hoses on the fuel oil day tank, fuel oil in open buckets, fuel oil on floor and foundations under engines and generators.

-- On February 25, 1982, welding operations were observed on diesel engines "A" and "C" with fuel oil still under all three emergency diesel generators. These examples are collectively considered a violation Item No. (322/82-04-13).

The licensee acknowledged these findings and committed to increasing surveillance and cleanup within buildings by providing 10 additional personnel for housekeeping and cleanup. On the morning of February 26, 1982, a tour of these areas by the inspector showed all the above findings concerning fire hazards and housekeeping had been corrected.

4.3.3 Emergency Diesel Generator Electrical Trip Lock-Out Features (IE Circular 77-16)

The circular described a situation where trip circuits supplied by a certain manufacturer were not bypassed in the emergency and fast-start modes (trip lock-out features). This resulted in an unexpected opening of the generator output breaker through a vendor-supplied field voltage sensing relay. A redundant relay supplied by the licensee had been bypassed; however, the vendor-supplied relay was not.

A review by the licensee of the Shoreham design for each diesel generator has shown that all protective relaying associated with the generator output breaker is bypassed in the emergency and fast-start modes, except the generator differential and generator overcurrent trips to the output breaker. This is consistent with the design requirements for Shoreham and meets the intent of the Circular. This Circular is closed.

4.4 ECCS Discharge Line Fill System

In order to make up possible leakage past check valves and maintain ECCS lines completely water-filled, loop level pumps, associated piping, valves, and instrumentation are provided. Two pumps service the core spray and LPCI systems - one pump for each division or loop.

The inspector verified that the installed configuration of the ECCS discharge line fill system conforms to the approved final design of Dwg. No. FM-20A-13 and related drawings. Pumps, valves, piping, instrumentation, supports and hangers were inspected by direct observation for the RHR Loop A system. These were determined to agree with the FSAR, flow diagrams and system descriptions. No discrepancies were noted.

4.5 Leakage Return System

The Leakage Return System (LRS) is a Nuclear Safety Related, QA Category I System whose purpose is to cope with a limited leak during a long-term post-accident period (SER, p.6-45). The system design is described in E&DCR's P-3299 through P-3299U. The installed system is designated a safety-related portion of system G11. The flow diagram appears in the E&DCR's and on drawing FM-46B. The piping was traced from the floor drain sump through the self priming pump mounted on a 41" high concrete pedestal to the junction with the Core Spray return line to the Suppression Pool through Penetration X10A. In addition, the piping run and pipe supports were compared with pipe Isometric Drawing IC 1546. The E&DCR's and drawings describe the installed system. The following observations were acknowledged by licensee staff:

- Caps on test, vent and drain lines were not completely installed as per drawing.
- PI-640C had been removed for work.
- Valve E21*03V-0021 had a nonconformance tag on it; the nonconformance was lack of an ASME code tag to be placed when certification was completed.
- The level control switch/level indicator *LE642C was installed in the TK-056C sump, but had not been electrically connected.

The inspector had no further questions regarding the LRS.

5. Management Controls

5.1 "As Built" Program

The inspector reviewed documentation and held discussions with cognizant licensee and A/E personnel to determine the adequacy of the licensee's program for revising and up-grading the drawings and other engineering design information to reflect the as-built/as-installed condition of the plant piping. The "as-built" program review by the inspector was carried out in conjunction with the physical inspection of the RHR and supporting systems. The inspector reviewed the licensee's approved and draft procedures (PP-38, CSI-9.14, IOC-63, IOC-63A) for the "as-built" program. The effectiveness of these procedures was assessed by reviewing the preliminary "as-built" drawings on a sampling basis. The inspector compared the information and data contained in the drawings to the physical layout and actual condition of portions of the system.

Responsibilities for the licensee's "as-built" program were ascertained from procedures and discussions with staff members. The inspector noted that the Project Engineer for Construction was responsible for preparation and submittal of "as-built" drawings and isometrics to the Site Engineering Office (SEO). The SEO Stress Engineer was responsible for reviewing, approving, and/or processing all "as-built" isometrics in accordance with SEO Memo No. 63A. In addition, he was to provide disposition and resolve any noncomformance identified by Field Quality Control (FQC) for all ASME piping or by construction forces for non-ASME, non-thermal, and non-seismic piping. For ASME piping, the FQC compared the isometrics to installed condition. For other piping, the Mechanical Superintendent of Construction was responsible for verification.

Furthermore, under the ASME B&PV code, the installer of ASME piping, Courter & Company, Inc., (holder of ASME 'N' stamp) is responsible for providing and certifying the drawing with the "as-installed" condition and other pertinent information. During the course of discussion, the inspector was informed that there was a procedure under development which would provide controls and assign responsibilities for identifying and resolving any conflict and/or disparity between the two parallel channels of "as-built" and "as-installed" information. The inspector was also informed that once the final "as-built" information was assembled and all conflicts resolved, the SEO would review the drawing to reconcile any stress problems in the "as-built" system configuration which might be in variance with the design as analyzed for stress, and certify the "as-built" system for the stress analysis.

Based on the above reviews and discussions, the inspector determined that the program to compile "as-built" information, to incorporate this information into the design drawings and isometrics, and to resolve any deviation from original design was still incomplete. This area remains unresolved pending further definition of this program, and formalization of control procedures to reconcile "as-built" and "as-installed" drawings and place the corrected, approved final drawing into the plant permanent record. This is Item No. (322/82-04-12).

The "as-built" program for electrical systems is comprised of three parts. One part is to verify and sketch the supports for cables and raceways and identify these on drawings for stress reconciliation; according to licensee representatives, this part is about 30% complete. The second part is to sketch the supports for conduits and relate these to drawings; this part is about 10% complete. The third part is the as-built sketching of conduits; this part is not yet started. No inspection of the electrical structural aspects of the "as-built" program was made because of the incomplete status.

5.2 Design Change and Noncomformance Control

5.2.1 Engineering and Design Coordination Report (E&DCRs)

The design and engineering changes of Shoreham site are primarily controlled by a system of Engineering and Design Coordination

Reports (E&DCR's). The responsibility for design and design control has been delegated by the licensee to the principal contractor and Architect/Engineer, Stone & Webster Engineering Corporation (S&W). The licensee's direct participation in design and its control was found to consist primarily of review of selected design changes, and audit of S&W design activities and controls.

Stone & Webster has a comprehensive system of design and design change control. These controls are adequately described in S&W's Engineering Assurance Manual (EAP). The inspector reviewed S&W procedure EAP 6.3, which controlled the initiation, problem resolution and distribution of E&DCRs. The E&DCRs were found to be the primary vehicle to initiate, resolve and/or implement changes to an approved design document. The inspector reviewed the status of several drawings and E&DCR's to assess the effectiveness of measures established for their adequacy, approval, currency of revision, and/or posting of changes. The major portion of this review was carried out in conjunction with the documents used for the "as-built" program inspection.

The inspector observed that the E&DCR system was also being used for documenting interpretations of design requirements, and site-project technical communications. This has led to a large number of E&DCR's. The inspector also noted that many E&DCRs themselves had been revised, modified, and/or augmented with additional information over a period of time without incorporation into the affected design documents. Furthermore, some E&DCRs, classified as "generic", were comprehensive primary design documents used over and over for a long period of time; these E&DCR's underwent many revisions themselves without incorporation in any drawing and/or specification. Such practices and uses of the E&DCR system, primarily intended for change control, has created a somewhat unwieldy and cumbersome system. The revision of primary design documents, i.e. drawings and specifications, has not kept pace with generation of E&DCRs. The drawings and specifications were posted with listings of E&DCR's affecting them. The inspector observed that, in the case of drawings FM-20A and FM-20B used for this inspection, the referenced E&DCRs numbered 26 and 21, respectively. These E&DCRs date back to June, 1978, for FM-20A and April, 1978, for FM-20B (E&DCR Nos. F-14071 and F-11993A, respectively). Because of the number and frequent revision of these documents, the design information and requirements were fragmented into numerous E&DCRs, drawings, and specification. This fragmentation makes it difficult to use drawings and specifications unless one is quite familiar with them and their pertinent E&DCR's. A clear, concise, and timely dissemination of technical and design information is fundamental to effective and error-free execution of engineered construction. The E&DCR system as implemented at Shoreham, with the lack of timely drawing revision, does not provide such dissemination. This is considered a weak area in the licensee's management control program and is identified as Item No. (322/82-04-14).

5.2.2 Non-conformance and Disposition Reports

The inspector reviewed a random selection of Nonconformance and Disposition Reports (N&D's) to assess the licensee's program for nonconformance control. The N&Ds were reviewed to determine the adequacy of nonconformance description, disposition, and controls over the implementation of N&D disposition. The review also considered any apparent evidence of repetitive nonconformance.

Based on this review and discussions with cognizant personnel, the inspector determined that the N&Ds contained sufficient details of the nonconformances to make informed judgement regarding the problem identified. The dispositions reviewed were technically proper and adequately detailed for the implementation of corrective actions. The N&D's were properly reviewed and approved by authorized personnel.

5.3 Proposed Technical Specifications

As part of the licensee's application for an operating license, proposed Technical Specifications (TS) were submitted in January, 1982. The inspector reviewed portions of these TS for the RHR and related systems to determine if the TS properly reflected the as-built plant and to determine if the proposed specifications were adequate to assure operability of the equipment. Two of the areas reviewed revealed the problems discussed below.

5.3.1 Snubber Table

Table 3.7.5-1 of the proposed TS lists safety related snubbers. The list was not accurate, in that:

-- Not all RHR System snubbers were included, e.g. E11-PSSP-807, 831 and 902.

-- The list did not recognize multiple snubbers under the same identifying number, e.g. E11-PSSP 824 has two snubbers.

-- The designation for "High Radiation Zone during Shutdown" and "Especially Difficult to Remove" snubbers did not appear reasonable. Apparently, 20 mrem per hour was used as a High Radiation Zone. It was not clear what guidelines were used to classify especially difficult to remove snubbers.

5.3.2 Plant Unique Features

10 CFR 50.36(b) requires that TS be submitted which are derived from the analysis and evaluations in the Safety Analysis Report. The inspector noted that important, safety-related, plant unique features described in the FSAR were not included in the proposed TS. The TS contained neither Limiting Conditions for Operation nor Surveillance Requirements for these plant unique systems: the RBCLCW System, RHR area coolers, LPCI Motor-Generator Sets, Drywell Floor Seal, Drywell Floor Seal Pressurization System, and the Leakage Return System. The inspector stated that a review of the FSAR to determine which additional systems should be included in the TS appeared appropriate. The discrepancies in the proposed Technical Specifications regarding safety related snubbers and the apparent omission of TS for plant unique systems are considered a weakness and are assigned Item No. (322/82-04-15).

5.4 Control of As-Built Information, Design Changes and Modifications Following System Turnover to Plant Staff

Configuration control up until a system is turned over from startup to the Plant Staff will continue to be documented on E&DCR's and Repair/Rework Requests. Following turnover to plant jurisdiction, this control will be accomplished in accordance with plant procedures.

The inspector reviewed these procedures and discussed their application with the Plant Manager and the Technical Support Manager. No safety systems have been turned over to plant jurisdiction as yet. Several small secondary plant systems have been turned over. A few minor modifications on these systems were made, using the approved procedures to maintain configuration control and to prove out the procedures. As a result of this and other experiences, the procedures are being revised to include flow charts and clear delineations of responsibility and to assure that the Training Department gets early information for incorporation into licensed operator training and that the Operating Department gets early information to trigger procedure changes. It is anticipated that revised procedures for design change and modifications will be approved by July, 1982. In addition, test engineers and plant operators have been instructed to make maximum use of existing plant operating procedures during startup testing to gain familiarity with them and to insure that they incorporate the as-built information.

Corporate engineering department procedures to describe the off-site engineering control of design changes and modifications are still under development. Procedures for the functioning of the Corporate Nuclear Review Board are also being developed. These procedures to cover off-site engineering and safety review functions are expected to be approved and in use within six months.

The inspector had no further questions regarding design change and modification following system turnover to the permanent plant.

5.5 Housekeeping

During the weeks of February 8-12 and 22-26, 1982, the inspection team observed the licensee's control and management of housekeeping, cleanliness, fire protection and fire prevention.

In the inspection of systems in the Reactor Building, the Primary Containment and the Screenwell Pump House, examples of many items of construction such as piping, valves, nuts, bolts, studs, hardware, steel rails, angle iron, I-beams, lumber, rags, paper and the like were observed in unwanted locations such as MOV housings and cable trays. Many of these items showed no evidence of have been moved in a period greater than the three weeks of February 8-26, 1982. Of particular concern were the many observed instances of dirt, grit and debris, in, on and around electrical boxes, cabling and operating shafts and stems of motor operated valves. On February 25, 1982, the inspector observed grinding being conducted next to an RHR pump with grit and dust settling around the unprotected RHR pump shaft. In the Screenwell Pump House, both rooms had excessive hardware and equipment not in use. A table was set up against electrical cabinet R24MCC1110 with hardware stacked on it. A ladder was set against electrical cabinet MCC-11B4. Both cabinet faces were labeled with NEC-OSHA signs stating "... area in front of electrical panels must be kept clear for thirty six inches in front of cabinet ...".

In the areas of the Fuel Oil Storage and Transfer (FOS&T) rooms and the EDG rooms, fire hazards included fuel oil in buckets and fuel oil drippings in these rooms. Debris was seen under the Emergency Diesel Generators (EDG's). These hazards were reported to licensee personnel on February 11th, but were not completely corrected until February 26, 1982. These housekeeping and fire protection observation form the basis for a Violation previously identified (322/82-04-13).

The requirements for housekeeping, cleanliness, fire protection and prevention are documented in the S&W Field Construction Manual, Section F parts I to VIII, which spells out fire protection, cleanliness, welding, inspections for prevention of fire; S&W Construction Site Instructions, Housekeeping No. 13.1 Revision 7, which spells out cleanliness requirements; and, ANSI N45.2.3, Housekeeping during Construction of Nuclear Power Plants. One statement reads "... local cleanup of contaminated areas is recommended as installation progresses, rather than one cleanup operation when installation is completed. Consideration should be given to sequencing installation and erection operations, when practical, to facilitate cleaning and cleanliness control ...".

These examples were judged a violation stemming from lack of management control for housekeeping, cleanliness and fire prevention.

Licensee management responded by correcting the fire hazards identified in the FOS&T and EDG rooms, confirmed by inspection February 26, 1982, and by committing to the assignment of a ten-person cleaning crew to plant buildings beginning March 1, 1982.

5.6 Cabinet Seismic Mounting in Control Room

The inspector reviewed FSAR requirements, instructions and drawings to determine whether the Standard Cabinets in the control room were installed in accordance with NRC requirements and FSAR commitments. FSAR Section 3.10.2.1.1B identifies five Standard Cabinets used in the design for determination of mounting bolt stresses and states that static seismic analysis was performed to verify that the mounting bolts of the standard cabinets are capable of withstanding the seismic environment. The five standard cabinets are:

-- Area Radiation Monitor, 236400(911)

-- TIP Control, 236X401 (913)

-- Startup Neutron Monitor, 236C402 (936)

-- Power Range Monitor, 236X403 (936)

-- Rod Position Information System, 236X404 (927)

Each cabinet was assumed to be floor mounted using 5/8-inch bolts in all mounting holes. Table 3.10.2B-2 gives necessary information for determining the safety factor of each cabinet and lists the assumed number of mounting bolts for each cabinet. Review of the factor of safety of each standard cabinet indicates that the mounting bolts for each cabinet are capable of withstanding seismic disturbances as specified in the Seismic Design Guide.

The inspector compared the number of mounting bolts actually used to that listed in Table 3.10.2B-2 for the Startup Neutron Monitor and the Power Range Monitor with the actual installation. Panel H11*PNL-635 and Panel H11*PNL-636 (Startup Neutron Monitor) were each installed with eight 5/8-inch bolts instead of the twelve bolts listed in Table 3.10.2B.2.

Panel H11*PNL-608 (Power Range Monitor) was installed with twenty 5/8-inch bolts. The table lists forty 5/8-inch bolts. This is a deviation from the FSAR and, in conjunction with the items in paragraph 3.1.3, is designated Item No. (322/82-04-01).

6. Unresolved Items

Unresolved items are matters about which more information is required to ascertain whether they are acceptable items, or whether violations or deviations. Unresolved items identified during the inspection are discussed in the report above in Sections 3.1.4, 3.2.7, 3.3.2, 3.3.4, 3.4.3 and 4.2.3.

7. Exit Interview

The inspection team met with the licensee representatives (denoted in Section 1) at the conclusion of the inspection on February 26, 1982 at the Shoreham construction site. The team leader summarized the scope of the inspection and discussed the inspection findings, including observations.

8. References

8.1 Drawings

FM-1A-13, Revision 13, Machinery Location Reactor Building

FM-1B-13, Revision 13, Machinery Location Reactor Building

FM-7A-3, Revision 3, Access between Buildings, Operating Floor

FM-7B-3, Revision 3, Access between Buildings, Ground Floor

FM-15A-12, Revision 12, 1P42, Reactor Building Closed Loop Cooling Water, Sh1

FM-15C-9, Revision 9, 1P42 Reactor Building Closed Loop Cooling Water, Sh2 & 3

FM-20A-13, Revision 13, 1E11, Residual Heat Removal, Sh1

FM-20B-13, Revision 13, 1E11, Residual Heat Removal, Sh2

FM-44A-10, Revision 10, 1M41, Fuel Oil Transfer

FM-44B-3, Revision 3, 1R43, Diesel Generator Air Start

FM-46B-8, Revision 8, 1G11, Radwaste Equipment & FDR Drains Reactor Building, SH2

FM-47A-11, Revision 11, 1P41, Service Water, Sh1

FM-47B-2, Revision 2, 1P41, Service Water, Sh2

FE-3C-9, Revision 9, Main Control Room Bench PNL 1H11 PNL 601, Sh3

FE-3E-7, Revision 7, Main Control Room Bench Bd. PNL 1H111 PNL 601, Sh5

FE-3QA-3, Revision 3, LPCI FDR Control 1R24 PNL 111 & 1R24 TRS-111X

FE-3QB-2, Revision 2, LPCI Control PNL 1R24 PNL 113A

FE-3QC-3, Revision 3, LPCI FDR Control 1R24 PNL 112 & 1R24 TRS 112Y

FE-3QD-2, Revision 2, LPCI Control PNL 1R24 PNL 113B

FE-9A-7, Revision 7, 480V EMERG SWGR BUS 111 SH1

FE-9E-5, Revision 5, 480V EMERG SWGR BUS 112, Sh2

FE-9MY-5, Revision 5, 480V MCC 1R24 MCC112Y

FE-9MZ-5, Revision 5, 480V MCC1R21 MCC111X - Reactor Building, Sh1

FE-12A-5, Revision 5, 480V Motor Operated Valves - Reactor Building Sh1

FE-12F-7, Revision 7, 480V Motor Operated Valves - Reactor Building Sh6

FE-12J-8, Revision 8, 480V Motor Operated Valves - Reactor Building Sh9

FE-12K-7, Revision 7, 480V Motor Operated Valves - Reactor Building Sh10

FC-28AB-3, Revision 3, Control Room & Diesel Generator Room Equipment FDN Details

FC-28D-1, Revision 1, Diesel Generator Room Ground Floor, Sh1

FC-28O-3, Revision 3, Control Room and Diesel Generator Room Misc. Conc. Details

BZ-8E-27-8, Revision 8, Sheets 1 to 3, Residual Heat Removal Piping, Pipe Supports

BZ-8E-28-8, Revision 8, Residual Heat Removal Piping - Pipe Supports

BZ-8E-38-5, Revision 5, Sheets 1 to 3, Residual Heat Removal Piping - Pipe Supports

BZ-8F-7-8, Revision 8, Sheets 1 to 5, Residual Heat Removal Piping - Pipe Supports

BZ-537H-58-1, Revision 1, Sheets 1 to 2, Control Room Station Vent Water Chiller Piping

Logic Diagram - RHR System, Drawing Nos. SK-25-7AAA thru SK-25-7AM, SK-25-7A thru SK-25-7Z

DC Elementary Diagram (4160V) - Drawing Nos. 5E1101 thru 5E1104

AC Elementary Diagram (120V) - Drawing Nos. 6E1146, 6E1147 and 6E1148

AC Elementary Diagram (480V) - Drawing Nos. 6E1105 thru 6E1145

Elementary Diagram - RHR Annunciators - Drawing No. 10ANB34

DC Elementary Diagram (125V) - Drawings Nos. 11E1102 and 11E1103

One Line Diagram - LPCI Loop Injection - FSAR Fig. 8.3.1-10 thru 8.3.1-12

Weld Map - Instrument Control Drawing Nos. 1E11-½K1004-1CN9-2 thru 1E11-½K1015-1CN9-2

Test Loop Diagram - RHR Drawing 1E11-001 thru 1E11-009.

Cable Block Diagram - E11 System Drawing Nos. (Stone & Webster File No.)

1.61-136D, 1.61-139D, 1.61-228A, 1.61-230A, 1.61-167, 1.61-172D, 1.61-202A, 1.61-219A thru 1.61-232A, 5E1101 thru 5E1104, 6E1101 thru 6E1148, 11E1101 thru 11E1103.

Cable Pull Tickets for Cable Nos.

1B31AGC204, 1B31BYC224, 1B31BYK222, 1E11AGC245, 1E11AGC262, 1E11AGC266, 1E11AGC422, 1E11AGK421, 1E11BYC251, 1E11BYC252, 1E11BYC272, 1E11BYC277, 1E11BYC432, 1E11BYK251, 1E11BYK431.

Instrumentation Tubing Drawings Nos.

PN-018-SK-001, PN-018-SK-002, PN-021-SK-001, PN-021-SK-002.

Equipment Location Drawing Nos.

FM-1A-11A, FK-1A-13, FK-1B-12, FK-1C-11, FK-1D-12, FK-1E-11.

8.2 Isometric Drawings

E11RHR E-2821-1C-12 Approved 1-30-82

E11RHR E-2821-1C-24 Approved 1-26-82

E11RHR E-2821-1C-25 Approved 1-6-82

E11RHR E-282-1-1C-33 Approved 2-5-82

E11RHR E-2821-1C-34 Approved 1-30-82

E11RHR E-2821-1C-47 Approved 1-18-82

E11RHR-E-2821-1C-69 Approved 1-18-82

E11RHR E-2821-1C-1105 Approved 1-8-82

Large Bore Piping Isometrics

3" DRW-24-151-2 IC-1546

16" WS-217-158-3 IC-138

16" WS-215-158-3 IC-139

Small Bore Piping Isometrics

P-33L9-1

P-33MO-1

P-33NI-1

P-33N2-1

8.3 Specifications

Specification No. SH1-056
Field Fabrication and Erection of Piping

Specification No. SH1-068
Design & Fabrication of Nuclear Power Plants
Piping Supports

Specification No. SH1-089
Diesel Generator Sets

Specification No. SH1-159
Electrical Installation

Specification No. SH1-224
Technical Requirements for Cleaning and Maintenance of Cleanliness For Installed Systems.

8.4 Bulletins, Circulars and Information Notices

8.4.1 Bulletins

USNRC IE Bulletin 81-03, April 10, 1981, "Flow Blockage of Cooling Water to Safety System Components by Asiatic Clams and Mussels."

USNRC IE Bulletin 79-08, April 14, 1981, "Events Relevant to Boiling Water Power Reactors Identified During Three Mile Island Accident".

8.4.2 Circulars

USNRC IE Circular 77-16, December 13, 1977, "Emergency Diesel Generator Electrical Trip Lock-Out Features."

USNRC IE Circular 78-19, December 29, 1978, "Manual Override (Bypass) of Safety System Actuation Signals".

8.4.3 Information Notices

USNRC IE Information Notice 81-21, 1981, "Potential Loss of Direct Access to Ultimate Heat Sink".

USNRC IE Information Notice 81-01, January 16, 1981, "Failure of General Electric Type HFA Relays".

8.5 Regulatory Guides, American National Standards and Institute of Electrical and Electronics Engineers Standards

USNRC Regulatory Guide 1.6, "Independence Between Redundant (Onsite) Power Sources and Between Their Distribution Systems".

USNRC Regulatory Guide 1.39, "Housekeeping Requirements for Water Cooled Nuclear Power Plants".

USNRC Regulatory Guide 1.62, "Manual Initiation of Protective Systems".

USNRC Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units used as Onsite Electric Power Systems at Nuclear Power Plants".

"American National Standard Guidelines on Fuel Oil Systems for Standby Diesel Generators". ANSI N195-1976.

"Criteria for Protection Systems for Nuclear Power Generating Stations". IEEE 279-1971.

"Criteria for Class 1E Power Systems for Nuclear Power Generating Stations".

"Qualifying Class 1E Equipment for Nuclear Power Generating Stations". IEEE 323-1974.

"Standard Criteria for Independence of Class 1E Equipment and Circuits". IEEE 384-1974.

8.6 Manuals and Codes

Stone and Webster, "Field Construction Manual, "Section F, Parts I through VIII.

Stone and Webster, "Site Instruction Manual", Section 13.1, Housekeeping.

DeLaval, "Diesel Generator Manuals", DSR-48 Volumes I, II and III.

Shoreham Nuclear Power Station, "Final Safety Analysis Report", Volumes 1 through 16.

Shoreham Nuclear Power Station, "Safety Evaluation Report", NUREG 0430 and NUREG 0420 Supplement 1.

National Fire Protection Association, "National Fire Codes - Codes and Standards".

8.7 Procedures

"Emergency Diesel Generator Electrical Preop Test". PT307.003A, Revision 0, Approved July 17, 1981.

"Emergency Diesel Generator Electrical Preop Test". PT307.003B, Revision 0, Approved July 17, 1981.

"Emergency Diesel Generator Electrical Preop Test". PT307.003C, Revision 0, Approved July 17, 1981

"As-built" Drawing Changes". PP-38, March 31, 1981.

"S&W Task-Large Bore As Building Procedure". CS1-9.14, February 2, 1982

"Procedure for Preparation and Review of As-Built Isometrics for Small Bore". SEO-63A, July 22, 1981.

"Preparation, Review, Approval and Control of E&DCR's." S&W EAP-6.3